ANALYSIS OF PAST, PRESENT AND FUTURE APPLICATIONS OF NUCLEAR POWER FOR PROPULSION OF MARINE VEHICLES

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ABSTRACT

The current status of nuclear marine propulsion systems is reviewed in an historical framework, with special emphasis on the following:

1) Types of reactor plants and their suitability for marine propulsion application,
2) Economic considerations involved in nuclear marine propulsion,
3) Inherent design requirements -- including radiation shielding, fission product containment and decay heat removal -- which make achievement of an economically competitive, commercial nuclear ship difficult, and
4) Advantages and prospects of a fleet of nuclear merchantmen to bolster the currently declining and non-competitive U.S. merchant marine.

This study is carried out by an extensive literature survey and correspondence and/or personal discussions with authorities in the fields of marine and nuclear engineering. Detailed descriptions are presented of the 4, non-military, nuclear ships built to date and of certain, improved, nuclear propulsion plant designs intended for application in the next generation of nuclear merchant ships to be built. Detailed background material regarding nuclear reactor engineering is also included.

Among the conclusions reached in this study are the following:

1) Economic competitiveness can be achieved for nuclear merchant ships by the use of high power, long routes, and high ship utilization; construction and operation of follow-on nuclear merchant ships now are necessary for near term achievement of this economic competitiveness.

2) The ship type presently most capable of achieving economic competitiveness is the high speed, nuclear-powered container ship operating on a long trade route equipped with quick-turnaround port facilities.

3) Construction and operation of a fleet of nuclear merchantmen may be the key to restoring the U.S. merchant marine to a position of economic competitiveness in the world maritime shipping industry.

4) The outlook for construction of a small fleet of second generation nuclear merchantmen in the mid-1970's is moderately optimistic; a necessary prerequisite is Government consensus regarding the rightful place of the U.S. merchant marine among the many other national priorities.

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I. INTRODUCTION

Just 17 years ago, nuclear power drove the USS NAUTILUS to sea, conclusively demonstrating to the world the technical feasibility of marine nuclear propulsion. Since then, over 200 more nuclear-propelled naval ships and 4 nuclear-propelled commercial ships have been built and operated; many others are being constructed.

Nuclear power has provided these ships with capabilities significantly greater than those of their conventionally powered counterparts -- but at a price. To date, none of the nuclear-propelled commercial ships that have been built has been economically competitive with their conventionally propelled counterparts.

The advantages of using nuclear energy to propel a ship are many. For naval ships, nuclear propulsion gives primarily the capability to steam at high speeds over long distances independent of land- and sea-based fuel oil supply points. For submarines, the nuclear plant's lack of need for air gives the added capability of remaining submerged for months at a time, greatly decreasing the submarine's detectability and vulnerability.

For merchant ships, nuclear propulsion gives greatly extended cruising ranges at high speeds without excessively large bunkering requirements; but more importantly, it gives the promise of significantly lower fuel cost and higher net earnings than a conventionally powered ship. Finally, the use

of nuclear propulsion may be the last remaining opportunity to halt the ominous decline of the U.S. merchant marine and to regain for it a competitive position among the world's maritime powers. Such a renaissance could give significant direct benefits to the United States both in peacetime and in the eventuality of international conflict.

The less common, more expensive materials and the more exacting standards of workmanship required to fabricate a nuclear plant result in propulsion plant capital costs considerably greater for a nuclear plant than for a conventional plant. In addition, because of the additional framing and shell plating reinforcement necessary to support the reactor plant and to protect it from collision and grounding, the nuclear ship itself (minus the propulsion plant) costs more to build than a conventional ship. Certain operating costs for a nuclear ship also tend to be considerably higher than for a conventional ship; e.g. nuclear liability insurance, amortization costs, and a larger shore staff due to special safety and engineering requirements. For a nuclear ship to be economically competitive, then, the nuclear fuel savings i.e. the difference in the costs of fossil fuel and nuclear fuel, must be sufficient to compensate for the higher nuclear capital and operating (other than fuel) costs.

Nuclear propulsion cannot at the present or in the foreseeable near future compete economically with conventional propulsion in all ship types operating on all trade routes.

Certain ship, route and even terminal characteristics are required for economic competitiveness of a nuclear propulsion plant today. These characteristics are all necessary to maximize the advantages of nuclear propulsion, and include: high shaft horsepower (SHP) and long trade routes, such that the conventional ship's bunkering requirements are large; port facilities capable of rapid cargo handling and quick ship turnaround so as to afford high ship cargo-carrying utilization; and high reliability of ship, men and equipment to support the high ship utilization necessary for competitiveness.

The very nature of the fission process, from which a nuclear propulsion plant derives its energy, gives rise to two fundamental problems which pervade the design of the nuclear plant; no similar or analogous problems exist in a conventional, fossil fueled, propulsion plant. These two problems are protection of the crew from the very high radiation levels resulting from reactor operation, and protection of both the crew and the general public from possible release of the prodigiously radioactive fission products under all conditions of operation and foreseeable credible accidents. It is primarily these problems which make attainment of low weight, small size, low cost, and the elusive, higher net earnings so difficult. Experience gained from building and operating the N.S. SAVANNAH has indicated more promising solutions for these problems than those used on SAVANNAH. Experience with follow-on nuclear merchantmen is expected to indicate

other areas wherein additional improvements can be made to enhance economic viability.

The purpose of this study is to collect pertinent information regarding the application of nuclear propulsion to marine vehicles, and to present this information in a logical and cohesive form so as to maximize its usefulness in understanding the many factors involved. The present and known future intended status of worldwide application of marine nuclear propulsion is reviewed and a forecast is made, in an historical perspective, regarding the likelihood of future nuclear vessels in the presently declining and non-competitive United States merchant fleet. Sources of information include available scientific, engineering and economics literature (all unclassified), as well as discussions with and correspondence from knowledgeable persons working in this field.

This study covers only nuclear marine propulsion; no consideration is given to nuclear power plants mounted on floating or submerged platforms for the purpose of supplying non-propulsion power.

II. BACKGROUND -- (ref's 1 through 17, 79, 86, 87, 89)

This section, which might be subtitled "Introduction to Nuclear Engineering," contains a brief general discussion of nuclear reactors. It is included to enable the reader with little background in nuclear reactors to better understand the material that follows.

A. TYPES OF REACTORS --

In chemical reactions, participating atoms form molecules different from those entering the reaction, while individually maintaining their original atomic identities. Since the only effect is a sharing or exchange of valence electrons, the nuclei of the participating atoms are unaffected. Each chemical reaction results in an energy release (or absorption) of up to a few electron volts (ev, equal to 1.519 x 10^{-26} Btu) and an accompanying ($E=mc^2$, courtesy of A. Einstein) loss (or gain) of mass, per ev energy change, of about 10^{-32} grams or 10^{-9} atomic mass units (amu). Since the amount of mass and energy exchange per chemical reaction is so very miniscule, obtaining of useful, large scale amounts of energy from chemical reactions requires the consumption of large quantities of reactants.

In nuclear reactions, participating atomic nuclei are transformed into nuclei quite different from those entering the reaction. The products of the reaction may be isotopes of the reactants, or completely different atomic nuclei. Each nuclear reaction results in an energy release (or absorption) of up to hundreds of millions of electron

volts (10^6 ev $\equiv$ Mev) and an accompanying loss (or gain) of mass, per Mev of energy change, of about 10^{-26} grams or 10^{-3} amu. Most nuclear reactions leave one or more of their products in a metastable, or excited, state. These products release their excess energy, usually by emission of nuclear radiation, at a later, random time. For large numbers of like excited nuclei, this release of excess energy follows an exponential decay ($\exp(-0.693t/T_{1/2})$, where t is the time from excitation and $T_{1/2}$ is the time period, characteristic for each particular nuclear species and excitation mode, required for half the excited nuclei present at the start of the period to release their excess energy).

The 2 classes of nuclear reactions of most importance for large scale energy production are nuclear fusion and nuclear fission. Fusion involves combining 2 or more light nuclei to form a single, heavier nucleus. Fission involves splitting a heavy nucleus into 2 or more separate, lighter nuclei. In both types of reaction a net loss of mass results in release of energy. The reason for this net reduction of mass is indicated in Figure II-1 below. Every existing nucleus has less mass than would be predicted by simply adding the masses of the Z individual protons and the (A-Z) individual neutrons that make up the nucleus; in such a calculation, atomic electron mass may be included with proton mass, or may be totally neglected with less than 0.05% error. This "mass defect" (mass defect = Z x (mass of 1 proton) + (A - Z) x (mass of 1 neutron) - (mass of the nucleus)) is a measure of

how tightly these nuclear particles are bound together, since an amount of "binding energy" exactly corresponding (via $E=mc^2$, as above) to this mass defect would have to be applied to the nucleus to separate it into its constituent particles.

Figure II-1 shows the average binding energy per nuclear particle (neutron or proton, collectively called nucleons) as a function of mass number, A. From this figure it can be seen that both the fusing of 2 or more light nuclei and the splitting of a heavy nucleus result in products with greater net binding energy per nucleon. This excess binding energy, on the order of 3 to 20 Mev per fusion reaction and 200 Mev per fission reaction, is released in the form of heat and nuclear radiations such as gamma rays, alpha particles (helium nuclei), beta particles (nuclear electrons) and neutrons. The complete fissioning of 1 pound of uranium, for example, produces about 3.5×10^{10} Btu of energy, or the same amount produced by combustion of 2,500,000 pounds of coal.

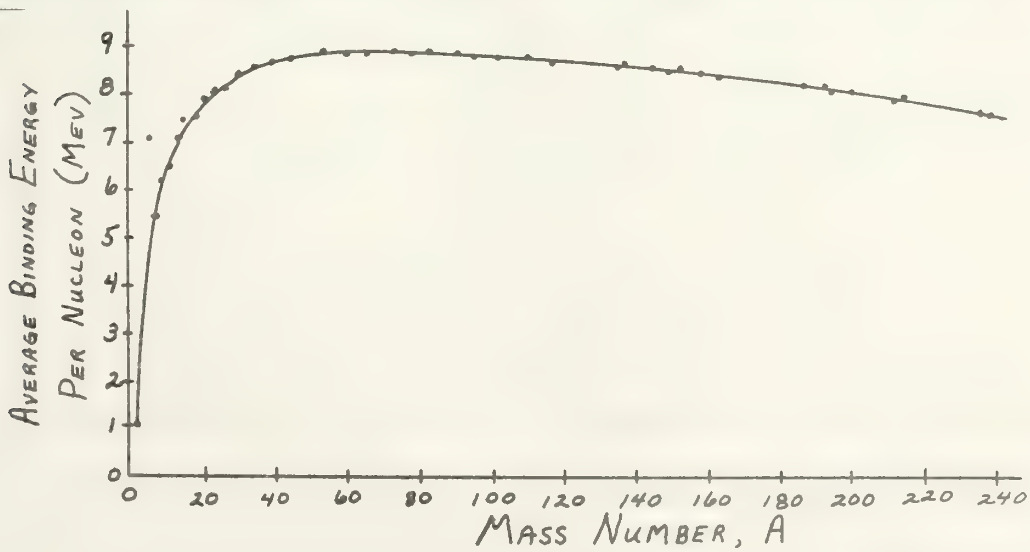

Figure II-1 Average Binding Energy per Nuclear Particle vs. Atomic Mass.

B. THE FISSION REACTION --

Since many, difficult technical problems remain to be solved before a man-made fusion reactor becomes a reality, the remainder of this discussion will be limited to fission reactors. Of the many, existing heavy nuclei, only a few can be fissioned; e.g., U-233 (Z=92, A=233), U-235, and Pu-239 (Z=94, A=239) can be fissioned by absorption of neutrons of all energies, while Th-232 (Z=90, A=232), U-238, and Pu-240 can be fissioned only by high energy neutrons, as will be explained below.

Since shortly after its initial discovery in 1939, the fissioning of a nucleus has been likened to the splitting of a drop of liquid. A certain threshold amount of energy must be given the drop in order for it to oscillate hard enough to overcome surface tension forces and split. As shown in Figure II-2, the excited drop passes through several oscillatory stages.

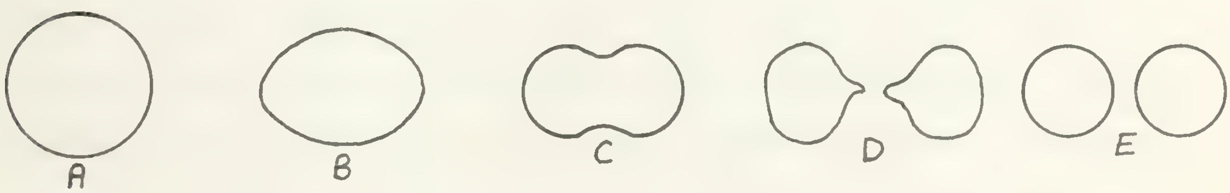

Figure II-2 Model of Liquid Drop Splitting

As the drop oscillates, it elongates from the spherical shape of A into an ellipsoid as in B. If at this point insufficient energy is available to overcome the surface tension forces, the drop will return to its original form. But if the deformation energy is sufficiently large, the drop

will rapidly pass through stages C and D, splitting into 2 droplets as in E.

In nuclear fission, a fissionable nucleus absorbing a neutron is excited by the binding energy of the new neutron in the compound nucleus (i.e. binding energy = ((mass of neutron) + (mass of original nucleus) - (mass of newly formed compound nucleus in de-excited, stable state) x c^2), plus whatever kinetic energy the neutron had when it entered the nucleus. If this total excitation energy is not sufficient to cause further deformation beyond B, the intranuclear forces will compel the compound nucleus to shed its excess energy, returning to a stable, spherical shape, by emission of particle and/or photon radiation from the nucleus. For large numbers of such insufficiently excited nuclei, this energy shedding will occur exponentially in time, as discussed above.

If the excitation energy is sufficient to overcome the intranuclear forces, however, the compound nucleus will rapidly split into 2 (or, very rarely, more) separate nuclei, giving off at the same time intense gamma radiation and an average of 2 to 3 neutrons with high kinetic energy. The excess energy and the large electrostatic repulsion forces between them cause these fission fragments to fly apart at very high speeds (3 to 5% that of light), imparting their roughly 170Mev kinetic energy to neighboring atoms (maximum fission fragment range is 0.010 in.).

The amount of excitation energy required to cause fission varies from 1 fissionable nuclide to the next, but a comparison of U-235 and U-238 will indicate why the energy of the absorbed neutron is important in the design and operation of a fission reactor. The threshold excitation energy required to fission U-235 is 6.5 Mev, while the difference in binding energies of the stable U-235 and U-236 nuclei is 6.8 Mev. Thus, absorption by U-235 of a neutron with zero kinetic energy produces a compound U-236 nucleus with an excitation energy of 0.3 Mev more than that needed to cause fission.

The threshold excitation energy of U-238, on the other hand, is 7.0 Mev, while the difference in binding energies of the stable U-238 and U-239 nuclei is only 5.9 Mev. Thus, absorption by U-238 of a neutron could not lead to fission unless the neutron brought with it a kinetic energy of at least 1.1 Mev.

Further understanding of the importance of neutron energy in the fission process would be enhanced by knowledge of the likelihood of a neutron's being absorbed by a nucleus, as a function of the neutron's kinetic energy. This likelihood of interaction is measured in terms of barns (1 barn = $10^{-24} cm^2$), i.e. an effective cross sectional "target" area presented by the nucleus to the moving neutron. The larger this area, the more likely it is that the neutron will interact with the nucleus. In a given interaction, one of several events can occur:

a) the neutron may be absorbed and fission may

result, as described above,

b) the neutron's absorption may only result in nuclear deexcitation by emission of radiation (so called "radiative capture"),

c) the neutron may be absorbed and re-emitted after giving up some of its energy to the nucleus (so called "nuclear inelastic scattering"), or

d) the neutron may simply bounce off the nucleus, without adding energy to the nucleus (so called "nuclear elastic scattering").

As will be seen below, each of these interactions has an important role in the operation of a fission reactor. There are 2 important generalizations that can be made at this point. One is that, for low energy neutrons, the cross section for absorption by most materials is proportional to 1/(neutron velocity), i.e. to the length of time the neutron spends in the vicinity of the nucleus. A consequence of this is that a low energy (in thermal equilibrium with its surroundings) neutron has on the order of 500 times more probability of fissioning a U-235 nucleus than a high energy (several Mev) neutron does. The other generalization is that significantly larger cross sections for neutron absorption may occur in several, discrete, narrow bands of neutron energies corresponding to favorable quantum mechanical energy levels (resonance energies) of the compound nucleus. Whenever a neutron is within such an energy band, the probability of its being absorbed is many times greater than when it has an

energy outside these bands.

C. FISSION REACTORS --

1. Reactivity and Multiplication Factor --

Since each fission reaction releases an average of 2 to 3 high energy neutrons, a steady state chain reaction of virtually any magnitude can be sustained if, on the average, one of these released neutrons can be made to cause another fission reaction. A fission reactor, then, must be constructed in such a configuration and of such special materials that, in spite of other (nonfission) reactions and in spite of neutrons leaking from the reactor, a steady rate of fissions can be maintained. This condition is conveniently expressed in terms of 2 quantities, the latter receiving more usage:

1) multiplication factor, k, defined as the ratio of the number of neutrons in any 1 "generation" to the number in the immediately preceding "generation" (a typical "lifetime" for a "generation" of neutrons, from "birth" at the moment of fission to "death" at the moment of low energy absorption or leakage out of the reactor core, is 0.001 sec), and

2) reactivity, ρ, simply defined as the ratio (k-1)/k.

Values of k greater or less than exactly unity (differing by even a very small amount), and values of ρ greater or less than exactly zero, correspond to an increasing or decreasing rate of fission. Since the amount of energy

produced per unit time is directly proportional to the number of fissions occurring in that time, k = 1.0000... and ρ = 0.0000... correspond to a steady state, constant level of power output. (3.1 x 10^{10} fissions/sec release 1 watt of thermal energy).

The amount of reactivity in a reactor core is determined by the relative extents to which the neutrons take part in 5 main processes:

1) complete loss or escape of the neutrons from the reactor, called leakage; participation in this process can be reduced by:

a) the presence of a neutron reflector around the reactor,

b) having a large volume-to-surface-area ratio, i.e. large overall dimensions for the reactor core (the part containing the fissionable material), and

c) reduction (by slowing neutrons down faster or absorbing them at higher energies) of the average distances neutrons travel between birth and death.

2) nonfission capture by fissionable nuclei, frequently called resonance capture since it is likely to occur mainly at resonance energies; participation in this process can be reduced a certain amount by minimizing the time spent by neutrons at resonance energies, but cannot be eliminated entirely since for a small percentage of the neutron absorptions in fuel (e.g. 16% for U-235) deexcitation of the compound nucleus will always take place by radiation

emission rather than by fission.

3) nonfission capture (often called parasitic capture) by nonfissionable nuclei (commonly termed "poisons") such as:

a) fission products,

b) reactor structural materials,

c) coolant used to remove heat,

d) cladding materials used to contain the fission products and to protect the fuel from chemical reaction with the coolant,

e) moderator used to slow down neutrons from roughly 1/10 the speed of light to thermal equilibrium in the reactor (a factor of 10^5 slower),

f) various impurities, and

g) fixed and moveable elements (e.g. control rods) used to control the fission rate;

participation in this process can be greatly influenced by selection and purity of the materials of which the reactor is comprised, and by the configuration of these materials relative to one another. The presence of such poisons imposes an upper limit on the amount of energy that can be gotten from the fuel, as further discussed below.

4) nonfission capture in fertile material in the core (such that the fertile material subsequently becomes fissionable due to radioactive decay, as discussed in more detail below.); participation in this process can be greatly influenced by the amount of fuel enrichment and/or the amount

of fertile material included in the core.

5) fission capture of neutrons by fissionable (fuel) nuclei; participation in this process can be increased by fuel isotope selection (e.g., Pu or U fuel system; amount of enrichment of uranium fuels in U-235, normally only 0.711% in natural uranium), and by reduction of participation in the above processes.

Reactivity control during reactor operation is achieved by: 1) operator action, such as by movement of poison-containing control rods or varying the concentration of poison (usually boric acid; boron is a very strong neutron absorber, producing lithium and helium), if any, in the coolant, and 2) inherent features of the reactor such as the following:

1) the Doppler effect -- Increase of fuel temperature increases the lattice vibrational energies of the fissionable nuclei, e.g. U-238, effectively increasing the width of the resonance capture energy bands and making it more likely that neutrons slowing down through these energy bands will be absorbed in nonfission captures; this effect is the primary mechanism for inherent prevention of rapid power excursions in uranium-fueled power reactors.

2) the temperature effect -- Increased temperatures reduce the densities of core materials, especially liquid coolants and moderators; since neutron interaction rates are proportional to n (nuclei/cm^3) x σ (cm^2 reaction cross section/nucleus) x ϕ (neutron flux, neutrons/cm^2sec), the more predom-

inant of 2 effects will determine whether the increased temperatures will yield a power reduction (a "negative temperature coefficient of reactivity"; inherently safe) or a power increase (a "positive temperature coefficient of reactivity"; not as safe.). These 2 effects are: a) reduction of n decreases the rate of neutron scattering and absorption, thereby increasing the distance a neutron travels while slowing down, increasing its time-average energy, and improving its chances of both leakage from the core and nonfission resonance capture; and b) reduction of n decreases the rate of parasitic absorption of neutrons by the coolant and the moderator, thereby increasing the chances of fission absorption by decreasing the likelihood of nonfission capture. This second effect would be especially important in dilute (high ratio of water to fuel) reactors. Neither effect is very important in fast reactors, which do not involve much neutron slowing down.

3) the void effect--Increase in the amount of voids in a reactor core, such as by boiling of some of the moderator or coolant, effectively decreases average nuclear density, thereby producing effects similar to the temperature effect; the result is increased time-average neutron energy, longer neutron slowing down time, increased nonfission resonance absorption, and reduced power (negative void coefficient of reactivity). For dilute reactors, the increase in voids can reduce parasitic captures in coolant and moderator enough to offset this effect and increase power (positive

void coefficient of reactivity).

2. Distribution of Power in the Core --

An important consideration affecting the power output of a reactor is the distribution of power generation in the reactor core. Ideally, one would like to have every element of the core producing a maximum amount of energy throughout its life; such a situation would represent an economically optimum fuel usage, since the fuel could be used until every portion of the core -- and not just the most limiting -- were expended to the desired burnup. (Buildup of fission product poisons and irradiation damage to fuel and cladding materials prevent complete fissioning of all the fuel in a reactor; the extent of fissioning is expressed in terms of fuel burnup, measured in Megawatt Days/ metric ton (1000 kg) of uranium, MWD/ tonne or MWD/T. A burnup of 9,000 MWD/tonne corresponds to fissioning about 1 in 100 fissionable atoms.) Unfortunately, this situation is a very difficult one to achieve in practice, mainly because the distribution of neutrons in a reactor core is far from uniform.

Power generation is proportional to n (fissionable nuclei/cm^3) x σ_f (cm^2 fission cross section/nucleus) x ϕ (neutron flux, neutrons/cm^2 sec). In an unreflected, cylindrical core with the dimensions shown in Figure II-3 the steady state (ρ = 0.0000...) neutron flux distribution, $\phi = \phi_o \cos \frac{\pi z}{He} J_o \left(2.404 \frac{r}{Re}\right)$, where ϕ_o is the maximum neutron

flux in the core and J_o is a Bessel function of zeroth order. R_e and H_e are extrapolated, effective dimensions (slightly larger than the actual physical dimensions) derived from neutron transport theory for the reactor materials involved; they mathematically account for the fact that the neutron flux is finite, not zero, at the physical boundaries of the core.

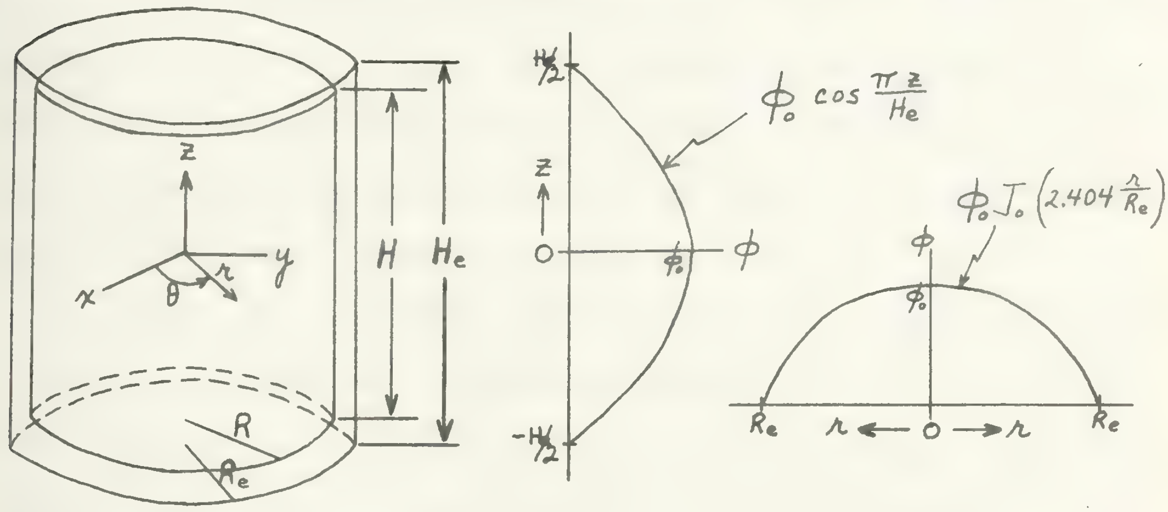

Figure II-3 Cylindrical, Unreflected Reactor Core Dimensions and Steady State Neutron Flux Distribution

The distribution of power generation in the core can be made more nearly uniform by application of 1 or more of the following schemes:

a) Increase flux near all core boundaries by surrounding the core with a reflector, a material with low neutron absorption and high neutron scattering cross sections, such as D_2O, graphite, beryllium, and H_2O; these materials reflect back into the core some of the neutrons that would

otherwise leak out near the edges of the core.

b) Reduce flux in the central portions of the core, either axially or radially or both, by permanent installation of discrete or distributed burnable poisons, such as boron, in concentrations decreasing from the core center to the edges.

c) Increase the product of n and σ_f , either axially or radially or both, by installation of larger fuel concentrations near core boundaries than near its center, and/or by increasing the enrichment of the fuel (e.g. in U-235) in the outer regions of the core.

d) Increase ϕ in the radially outer core regions in thermal reactors with negative temperature coefficient by having the return (cooler) coolant flow through these regions first, become heated, and then flow through the central core region.

e) Increase the number of fuel elements that operate at or near fuel centerline and cladding temperature upper limits by installing coolant flow orifices at the core entrance or exit, or both, such that the amount of coolant to cool each fuel element decreases radially outward.

f) Increase the effective length of the fuel rods, thereby flattening the axial distribution of ϕ, by operating with a burnable poison such as boric acid dissolved in the coolant so that control rods can be more fully withdrawn from the core.

3. Decay Heat Generation after Reactor Shutdown --

Shutting down a reactor, unlike a fossil-fired boiler, does not abruptly stop the generation of power; excited fission fragments continue to decay after reactor shutdown, each at its own individual characteristic rate, releasing significant quantities of energy as they decay. Figure II-4 shows the ratio of decay power, P_S, to power before shutdown, P_O, for various reactor operating times before shutdown, θ_o . To prevent severe damage to the reactor core, prodigious contamination of the primary coolant system, and possible release of fission fragments, it is essential that this decay heat be removed as it is produced. This can be done by use of normal coolant pumps, installed emergency pumps, natural circulation, or, in the event of loss of coolant such as by rupture of the coolant boundary, by emergency injection of additional coolant into the core.

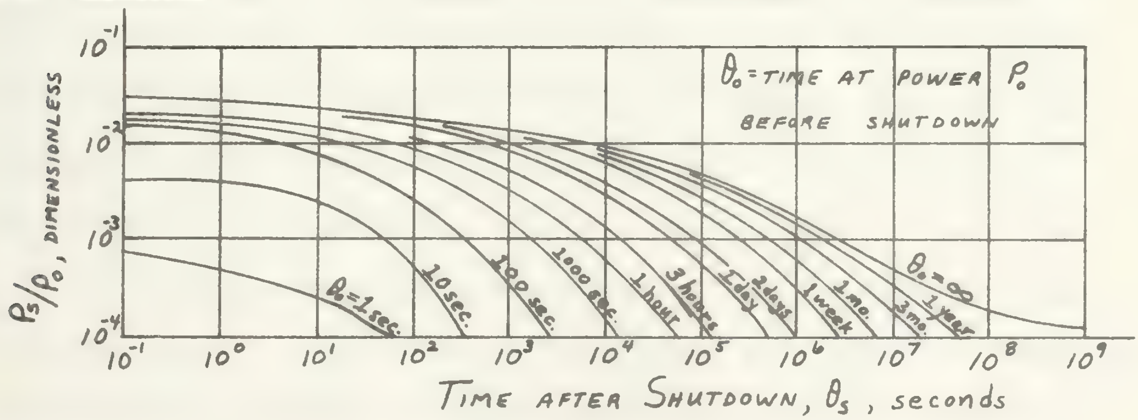

Figure II-4 Ratio of Power after Shutdown to Power before Shutdown for Various Prior Periods of Reactor Operation

4. Radiation Shielding --

Since the fission process itself, as well as decay of fission products and excited compound nuclei formed by neutron absorption, releases intense, penetrating and potentially lethal radiation, an inherent part of a reactor around which people must work and live is a radiation shield. For shielding purposes, radiation resulting from reactor operation can be conveniently grouped into 2 separate categories:

1) neutron radiation, which emanates only from the core; and

2) gamma radiation, which emanates from the core, the primary coolant, and reactor plant materials (such as the reactor pressure vessel which contains the core) which have captured a neutron from the core.

Other radiation either needs no shielding other than the material containing the primary coolant, because it is not penetrating (e.g., alpha and beta), or needs no shielding because it is so penetrating that it is not absorbed by people (e.g., neutrinos).

The primary shield, placed close to the core to minimize weight and volume, is designed primarily to attenuate neutrons for personnel protection during reactor operation and so that neutron activation gamma radiation dose rates outside this shield are acceptable. Low Z (preferably hydrogenous, such as water or polyethylene) material is used to slow down neutrons from about 0.1 Mev to absorption, since

at these energies a neutron will lose on the average 1/2 its energy by elastic collision with a H nucleus and very little of its energy by elastic collision with a high Z nucleus. High Z material, such as lead or iron, is used on the core side of, or interspersed with, this hydrogenous material to slow down neutrons by inelastic collision from fission energies (typically 5 to 10 Mev) to below about 0.1 Mev. This shield must also contain sufficient high Z material to adequately attenuate fission product and neutron activation gamma radiation following reactor shutdown to allow personnel access to reactor plant equipment outside the reactor pressure vessel, but inside the secondary shield, for routine inspection, preventive maintenance, and repair.

The secondary shield is more extensive than the primary shield, enveloping not only the reactor core, but also the entire primary coolant piping system and all auxiliary systems which contain radioactive coolant. The secondary shield is designed to attenuate both the neutron and gamma radiation which gets through the primary shield during reactor operation and the intense gamma radiation from the coolant (e.g., for water cooled reactors, 6.13 and 7.1 Mev gamma rays from the reaction 0-16 (n,p) N-16* $\xrightarrow[T_{1/2} = 7.3 \text{ sec.}]{}$ 0-16 + β^- + γ). Radiation levels outside this shield must be low enough to permit routine watchstanding and maintenance work to be performed in adjacent compartments during reactor operation without exposing personnel in excess of biologically safe, federally prescribed limits. Localized shielding installed

around gamma radiation sources such as coolant purification system filters and/or demineralizers is often included in the category of secondary shielding.

To minimize the weight and volume of the entire shield complex, thereby reducing the cost of the plant and of the ship, shielding optimization studies are performed as part of the plant design. Typical input variables to these studies include shield material physical and nuclear properties, costs and configurations, with specified sizing ranges for evaluation. Typical output variables are shield weights, volumes and costs. Optimized shipboard shielding designs typically have proportions such as in the following table:

	Shield Weight	Shield Volume	Shield Surface Area
Primary Shield	25-30%	30%	20%
Secondary Shield	70-75%	70%	80%
Materials primarily for neutron attenuation	10-20%	80%	-----
Materials primarily for gamma attenuation	80-90%	20%	-----

The shield complex must be designed adequately to withstand shipboard shock and vibration and to dissipate the heat deposited in it by the radiation without undue stresses.

Radiation dose to personnel is measured in a unit designated a rem, for roentgen equivalent man. Personnel

exposure in rem is equal to whole body exposure in roentgens times a factor called RBE, for relative biological effect. RBE values are as follows: gamma - 1, thermal neutrons - 3 to 5, fast neutrons - 10, and alpha - 20. A roentgen is a radiation dose resulting from exposure to that amount of less-than-3-Mev x or gamma radiation that causes an energy deposition of 93 ergs per gram of human tissue. Putting these somewhat hard-to-understand terms into a more meaningful framework, 600 rem or more dosage is fatal without extreme medical measures; 400 rem is fatal within 60 days for approximately half of those who receive it. A dosage of 50 rem produces only a transient effect in white blood cell count with no known, clinically demonstrable, permanent effects. The effects of dosages less than 50 rem are very difficult to detect or predict, especially as regards genetic effects (which have a lower threshold than somatic effects). Much deliberation by learned men, however, has resulted in the radiation exposure limits delineated in Title 10, Article 20, U.S. Code of Federal Regulations. Basically, these limits permit no more total cumulative whole body dosage than 5(N-18) rem, where N = person's age in years (occupational exposure to radiation before 18 years of age is not permitted) and no more than 3 rem in any 13 consecutive weeks; practically, this results in an administrative maximum allowable whole body dose per week of between 100 and 300 millirem (mrem = 0.001 rem).

5. Types of Fission Reactors --

Reactors can be classified in many different ways. Some of these ways are as follows:

a. By the neutron energies at which most of the fissions occur:

1) Fast reactors if most fissions are induced by neutrons with over approximately 0.1 Mev energy.

2) Intermediate, or epithermal, reactors for neutrons between about 0.1 ev and 0.1 Mev.

3) Thermal reactors for neutrons with less than about 0.1 ev energy.

b. By the type fuel used (can be further subdivided into ceramic (e.g. UO_2 or UC) or metallic (e.g. U and its alloys) forms):

1) Natural uranium, containing 0.711 w/o U-235.

2) Enriched uranium, containing more than 0.711 w/o U-235.

3) Pu-239.

4) U-233.

5) Mixtures, especially for breeding (creating fissionable from nonfissionable isotopes), such as PuO_2 - UO_2 or UC - ThC.

c. By type of moderator used:

1) H_2O

2) Graphite (carbon)

3) D_2O

4) Beryllium or beryllium oxide

5) Organic liquids, such as terphenyl

d. By the type of coolant used:

1) Liquids such as H_2O, D_2O, or organic fluids.

2) Gases such as helium, CO_2, air, or hydrogen.

3) Liquid metals such as Na, K, NaK, or Li.

Not all of the above classifications are independent. For example, a water-cooled and -moderated reactor is necessarily a thermal reactor. While many combinations of design variables such as those listed above are technically feasible, only a relatively few stand out as suitable, economically and otherwise, for large scale power production. These are discussed individually below.

D. TYPES OF FISSION REACTOR POWER PLANTS --

1. Pressurized Water Reactors (PWR) --

The PWR, shown schematically in Figure II-5, is currently the most widely used and well developed reactor type for power production. Water serves the multiple function of coolant, moderator and reflector; an excellent heat transfer agent, it is safe, inexpensive, easy to handle, and plentiful, and has well-known physical and thermodynamic properties. The primary coolant is maintained at pressures around 1500-2000 psia, permitting core outlet temperatures of the coolant

around 550-600F without bulk boiling. Steam conditions at the power turbine are typically 700-1000 psia at 500-550F, giving plant efficiencies of up to 30-32%. The high primary system design pressures of 2,000-2,500 psia necessitate a thick, heavy and costly reactor pressure vessel, a significant disadvantage. Even higher primary coolant pressures could be used, but would only result in complications in pressure vessel design and greater cost, with little increase in coolant temperatures and, thereby, plant efficiency.

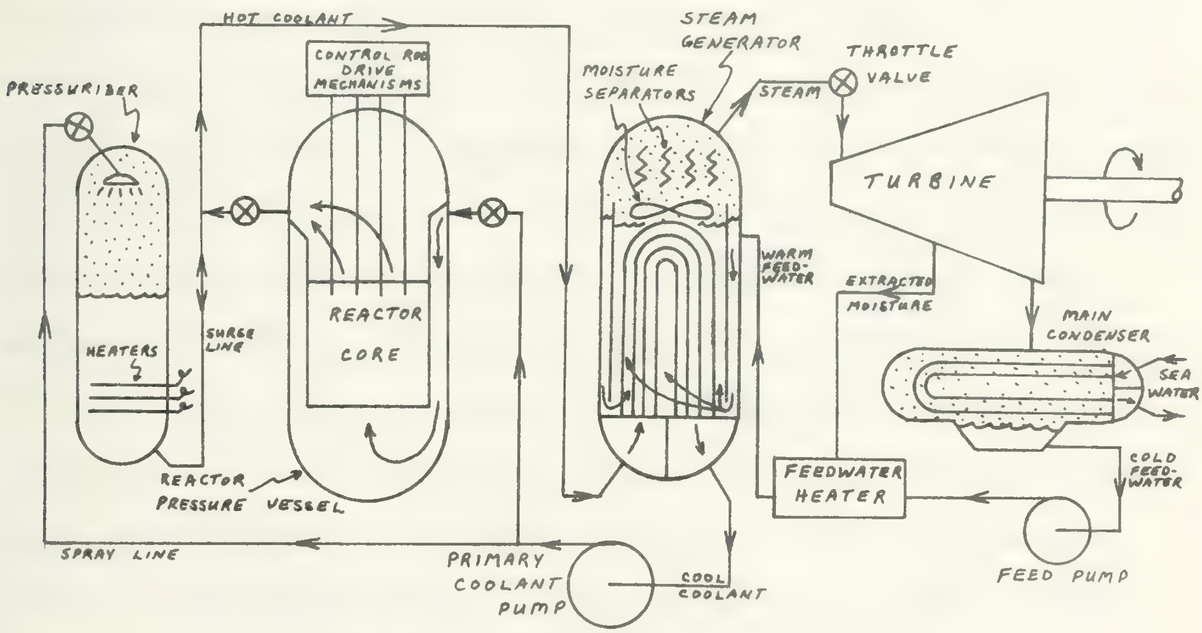

Figure II-5 Simple Flow Diagram of a Pressurized Water Reactor Power Plant

Careful selection of materials and rigid coolant chemistry control are vital to the successful operation of a PWR, not only from the standpoint of minimizing parasitic absorption of neutrons but to ensure long plant life in spite of the high propensity of high temperature water to corrode

most metals. The choice of reactor core materials is rather narrow; zirconium alloys and stainless steels appear to be the most suitable; for marine usage zirconium alloys have the advantage of providing excellent long term integrity of fuel elements in the presence of sea water thus minimizing the possibility of release of radioactive products in the event the ship sinks.

Coolant pH must be kept in a narrow range suited to the materials used, while dissolved oxygen, chloride ions, and soluble salts, especially of Cu, Cd, Co, Au, Pb and Ag, must be kept in extremely low concentrations in order to keep the corrosion rates of materials in contact with the coolant down to acceptable levels. This necessitates periodic coolant sampling and chemical analysis, followed by addition of necessary chemicals for pH control and hydrogen for oxygen recombination, plus continuous circulation of a fraction of the coolant through filters and demineralizers for dissolved salt and corrosion and wear product removal. Failure to remove impurities such as corrosion and wear products will result in their radioactivation as they pass repeatedly through the core. Their subsequent deposition on core or primary coolant flow surfaces can lead to unacceptable fouling of heat transfer surfaces (fuel elements can overheat and burn out, releasing fission fragments to the coolant) or unacceptably high radiation levels inside the secondary shield (routine inspection and maintenance can become prohibitively expensive and time-consuming).

The fuel system most commonly used in PWR's is uranium dioxide, a ceramic. UO_2 has excellent corrosion resistance in water (a desirable quality in case of cladding failure), good resistance to radiation damage, and a small cross section for parasitic neutron absorption. Because of UO_2's poor structural strength, however, cladding is required not only to keep fission fragments out of the coolant, but to provide necessary structural support for the fuel. U-235 enrichments greater than the natural 0.711 w/o are needed to sustain a chain reaction with ordinary, light water as the moderator. If heavy water (D_2O, normally 1 part in 7,000 in ordinary water) were used for the moderator, natural uranium fuels could be used; although the extra cost of fuel enrichment would thereby be avoided, an additional expense for D_2O ($>$ \$30/lbm) would be incurred. Which moderator is more economical depends on many factors; while Canada, for example, is widely using the more-economical-for-them D_2O-moderated PWR, the U.S. with readily available enriched fuel and higher carrying charges on D_2O finds the H_2O-moderated PWR a more economical power producer. Another significant advantage of ceramic fuels (e.g., UO_2, UC) as compared with metallic fuels is a maximum fuel burnup capability some 10 times greater (30,000 MWD/tonne vs. 3,000 MWD/tonne). Metallic fuels tend to swell and grow excessively if used beyond this burnup limit, due to their low temperature (e.g. 1235F for uranium) phase change and fission gas diffusion, collection in pockets, and expansion.

The very low compressibility and high coefficient of thermal expansion of water would cause severe pressure changes due to changes in coolant temperature associated with normal load changes, were a surge chamber not provided to accommodate coolant volume changes. The pressurizer provides such a chamber. Essentially a small boiler heated by electric immersion heaters to a temperature corresponding to saturation for the desired primary coolant operating pressure, the pressurizer is kept about half full of water and half full of steam. The bottom is connected via a surge line to a hot (reactor outlet) leg of the primary coolant piping and the top via a spray line to a cold (reactor inlet) leg; valving in the spray line controls the flow rate of coolant into the steam dome. Automatic intermittent heater and spray operation maintain primary pressure within the desired range. Pressurizer internal volume is typically 20-25% that of the reactor pressure vessel.

Steam for driving the power turbines is formed in the steam generators, usually recirculation type, vertical or horizontal tube-and-shell-type heat exchangers. Primary coolant heated in the core is pumped through thousands of parallel tubes in the steam generators where it transfers heat to the secondary working fluid. Adequate steam quality ($>$99.75% typically) is assured by installation of moisture separators, typically swirl vane and vane type, in each steam generator between the tubes around which boiling occurs and

the steam outlet. The steam is used in a Rankine cycle with moisture separation in and between high- and low-pressure turbines to reduce turbine blade erosion and increase turbine efficiency, and with extraction feedwater heating incorporated to improve thermal efficiency of the cycle. This cycle is similar to that used in fossil-fueled steam propulsion plants.

The PWR has been used extensively for land-based electrical power generation since 1954 in the U.S.S.R. and since 1957 in the U.S. It has propelled over 200 naval vessels, both submarines and surface ships, since the USS NAUTILUS went to sea in 1955, and has been used in all nuclear propelled merchant vessels constructed to date: NS LENIN (U.S.S.R.), NS SAVANNAH (U.S.), NS OTTO HAHN (W. Germany), and NS MUTSU (Japan). Details of these merchant vessels are provided in Appendix I.

2. Boiling Water Reactor (BWR) --

The BWR, shown schematically in Figure II-6, represents the most direct practical means of converting nuclear energy into useful power on a large scale. As in the PWR, water has the multiple function of coolant, moderator and reflector. Unlike in the PWR, the BWR coolant boils in the core. Entrained moisture is removed either inside the reactor pressure vessel or in a separate vessel located on a level above the reactor pressure vessel; the steam is fed directly to the power turbines. Removed moisture is mixed with the feedwater and recirculated to the core inlet through

downcomers by the recirculation pumps. Forced recirculation is necessary in the BWR to avoid unstable and erratic coolant flow through the core due to chugging oscillations caused by bulk boiling and resonant U-tube type interactions between the coolant in adjacent flow channels. Such oscillations also limit attainable core power densities and require the BWR reactor pressure vessel to be 25 to 50% larger than that for a PWR of the same power output. This disadvantage is partially offset, though, by the lower operating and design pressures of the BWR (about half those of the PWR), which permit use of thinner wall, more easily fabricated, less expensive primary components, and by the lack of a need for steam generators or a pressurizer (pressure is maintained by the buffering action of the steam dome pressure on the coolant).

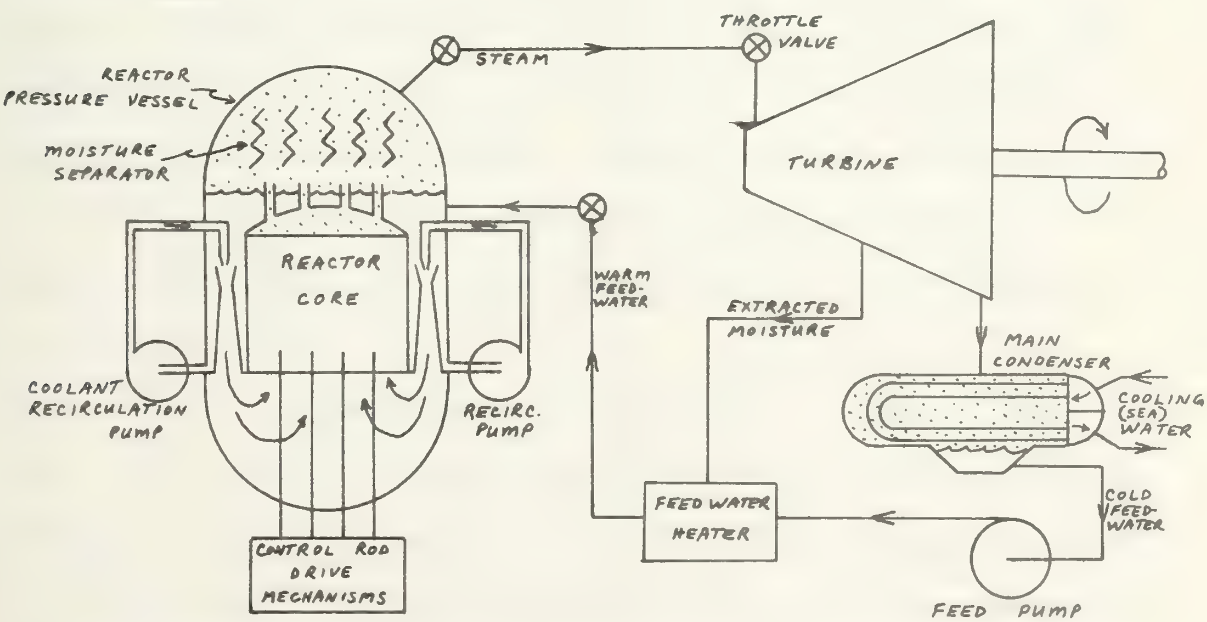

Figure II-6 Simple Flow Diagram of a Boiling Water Reactor Power Plant (Direct Cycle)

Presence of large amounts of steam bubbles in the upper portions of the coolant flow channels in the core adds large amounts of negative reactivity to the upper core (via the void effect discussed above), causing neutron flux to be peaked in the lower portions of the core. To avoid aggravating this already undesirable nonuniformity of neutron flux, it is necessary to withdraw control rods downward such that the length of rods remaining in the core tends to reduce neutron flux in the lower, rather than in the upper, portions of the core. To meet this requirement, mechanical design simplification usually dictates mounting the control rod drive mechanisms beneath the reactor pressure vessel. This feature tends to make the BWR less suitable for shipboard installation, in which the reactor pressure vessel (a heavy weight) would have to be mounted relatively high in the vessel and the control rod drive mechanisms would have to be exposed to the danger of damage in the event the ship runs aground. Another potential difficulty in marine use of a BWR plant is the reactivity changes associated with void movement and volume change due to ship motions caused by shock, impact and seaway forces.

The direct use of primary coolant as the steam plant working fluid demands more stringent controls over primary coolant chemistry. Although completely pure coolant and long steam lines would theoretically produce, downstream of the turbine throttle valves, only small gamma radiation levels during operation and no radiation levels after reactor

shutdown, even a few ppm mineral content in the coolant can cause long-lived, high radiation levels in secondary plant piping and components, requiring additional shielding. Accordingly, much more attention to primary chemistry control and to secondary equipment design is necessary than for a PWR plant. Pockets and crevices that may collect and retain radioactive particles must be scrupulously avoided in piping and component design. Pockets necessitated by other design requirements must have built-in drainage provisions, and turbines must have provisions for filling (and draining) with decontaminating fluids when necessary and before dismantling for maintenance.

A major drawback of the direct cycle BWR just described is that the reactor is not load-following; i.e., an increase in steam flow demand to the turbines decreases core pressure and increases core void fraction, resulting in decreased reactor power generation, and vice versa for decreased steam flow demand. Because of this characteristic of the direct cycle BWR, even small changes in steam flow demand must be accompanied by control rod motion to offset pressure and void reactivity additions and ensure a reactor power level corresponding to steam demand.

To overcome this drawback, more recent BWR plants have been built with variable speed recirculation pumps. The variable recirculation flow direct cycle BWR plant effects load following in the following manner. Increased steam demand is sensed by automatic circuitry which increases the

recirculation flow rate by a programmed amount, reducing the void fraction in the core. The enhanced neutron moderation and reduced leakage increase reactor power (via negative void coefficient of reactivity) until the void fraction increases to its original value, making reactor power and steam demand equal. The inverse occurs for decreased steam demand.

3. Gas Cooled Reactors --

Of several types of gas cooled reactors that are technically feasible, the indirect closed cycle gas steam type using thermal reactors makes up the majority of gas cooled reactors being built or developed in the world today. This type plant is shown schematically in Figure II-7. Gases such as CO_2, He and air are safe, relatively easy to handle, have low cross sections for parasitic neutron absorption, and are -- except for helium -- plentiful and cheap; they also may be operated at high temperatures without requiring high pressures. Disadvantages of gases for coolants are their low heat transfer and heat transport characteristics (requiring large heat transfer surface areas and flow passages) and their high pumping power requirements(typically consuming 8 to 20% of plant gross power). Design optimization for maximum thermal efficiency usually leads to operating fuel elements at as high temperatures as metallurgically permissible and obtaining a high gas temperature rise across the core by use of a low gas mass-flow rate and pressurization of the gas. Keeping reactor and steam generator sizes below those of a PWR also requires high gas pressures and use of finned heat transfer surfaces.

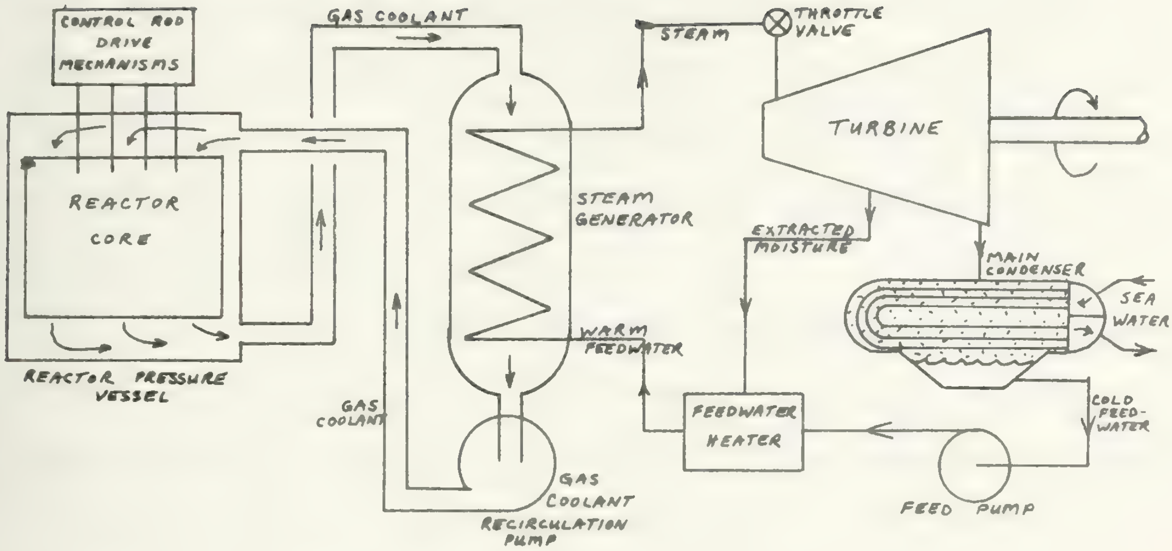

Figure II-7 Simple Flow Diagram of a Gas Cooled Reactor (Indirect Closed Cycle Gas Steam Type)

Gas coolants most often used are CO_2 and He. Other gases are less desirable due to one or more reasons such as the following: too expensive (e.g. Ne); chemically corrosive (e.g. O_2, H_2); unstable under irradiation (e.g. NH_3); high cross section for parasitic neutron capture (e.g. N_2); and unacceptably high induced radioactivity (e.g. Ar). The moderator most commonly used in gas cooled reactors is graphite; this tends to make gas cooled cores rather large. Slightly enriched uranium ceramic fuels (UO_2, UC, etc.) are commonly used. The enrichment allows use of higher temperature cladding materials such as stainless steel (more highly neutron-absorbing than lower temperature zirconium alloys), and allows use of smaller fuel elements and a smaller reactor core. The ceramic permits higher temperature operation of

the fuel. Typical gas pressures used range from 200 to 700 psia; core inlet gas temperatures range from 500 to 700F and outlet temperatures from 700 to 1400F. Typical steam conditions and thermal efficiencies range from those of a PWR plant to 2400 psia, 970F and 39.4% efficiency, comparable to current fossil-fueled plants.

A unique feature of most gas cooled reactors is the use of prestressed concrete reactor pressure vessels. These vessels can be cheaply built on site in almost any size for pressures up to about 1250 psia, whereas steel vessels of comparable sizes would be difficult and expensive to fabricate in the required thicknesses and would have to be field welded. Use of such concrete vessels allows the entire primary coolant system to be contained inside the vessel, reducing the danger of primary system boundary rupture and loss of coolant. Since the occurrence of such an accident at high power (this also applies to loss of coolant flow at high power) could cause melting of fuel and cladding, its possibility strongly indicates the need for development and installation of a reliable emergency core cooling system. Another potential hazard in the use of gas cooled reactors for ship propulsion is that introduction of sea water into the core (e.g., from sinking of the ship and breaching of the primary coolant boundary) could result in an uncontrolled nuclear excursion. Design of the core to preclude this possibility involves complications such as installation of an excessively large number of large control rods with a

fast-acting, very reliable scram capability. Gas cooled reactors have been successfully used for land-based power generation since 1957.

Conservation of the world's limited uranium supply for the use of future generations dictates development of breeder reactors, which can convert certain nonfissionable materials (termed "fertile") into fissionable fuels by neutron absorption and subsequent decay. Examples are Th-232 (fertile) $+ {}_0n^1$ (any energy) $\longrightarrow \gamma$ + Th-233 $\xrightarrow[T_{1/2} = 22\frac{1}{2}\,min]{} \beta^-$ + Pa-233 $\xrightarrow[T_{1/2} = 27\,days]{}$ β^- + U-233 (fissionable), and U-238 (fertile) $+ {}_0n^1$(fast only) $\longrightarrow \gamma$ + U-239 $\xrightarrow[T_{1/2} = 24\,min]{} \beta^-$ + Np-239 $\xrightarrow[T_{1/2} = 2.3\,days]{} \beta^-$ + Pu-239 (fissionable). Although it is unlikely that a breeder reactor would be used to propel a ship, since it would be significantly larger and heavier than a non-breeding reactor, the subject is included here for completeness. The 2 reactor types currently being developed for breeding are the gas cooled fast reactor and the liquid metal cooled fast reactor discussed below. Fast reactors are used to avoid the neutron losses due to parasitic absorption and leakage at intermediate and low energies, thereby enabling utilization of a larger percentage of neutrons for the breeding reaction. Hydrogenous moderators, of course, cannot be used in fast reactors because of their highly moderating characteristics.

4. <u>Liquid Metal Cooled Reactors</u> --

The liquid metal cooled reactor, shown schematically in Figure II-8, can be a thermal (if a separate

moderator is used), an intermediate, or a fast reactor, depending on the amount of neutron moderation permitted. In general, liquid metals such as Na, NaK, K, Hg, Bi and Pb have low moderating capabilities, low cross sections for fast neutron absorption, and excellent heat transfer and fluid flow characteristics, making them particularly suitable for fast reactor coolants. Their relatively low melting points (typically 150-500F) and low vapor pressures make possible high core outlet temperatures with low coolant pressures, resulting in high thermal efficiencies (up to 40%), smaller cores and less expensive reactor pressure vessels. Their high thermal conductivities result in reduced hot spot factors and lower temperature gradients in the core, reducing the probability of core structural warpage. As shown in Table II-1 below, coolant pumping power is only slightly greater than that required for PWR plants; mechanical or electromagnetic pumps can be used.

Table II-1 also compares the values of convective heat transfer coefficients for various reactor coolants. From the standpoint of only pumping power and heat removal, it can be seen that water is superior to the other coolants, followed by liquid metals, the organics, and finally by gases. Typical coolant temperatures are 900F at core inlet and 1100F at core outlet, with coolant pressure slightly above atmospheric, giving a steam temperature of about 1050F.

Disadvantages of liquid metals include:

1) high induced radioactivities with long half lives

(e.g., K-41: 12.4 hrs; Na-24: 15 hrs; and Hg: 44 min to 47.9 days); this requires heavy shielding (more than offsetting the weight savings associated with core compactness), long waits before access after shutdown, and use of an intermediate heat exchange loop to isolate the radioactive coolant from the secondary plant working fluid (coolant radioactivity levels are several thousand times those for water),

2) high chemical activity with air and water; this necessitates the intermediate heat exchange loop, inert atmospheres and/or very tight coolant boundaries,

3) necessity to keep molten at all times; this requires use of external heaters when shutdown,

4) necessity for more careful design to avoid thermal shock, and

5) relatively high expense.

Table II-1 Comparison of Convective Heat Transfer Coefficient and Relative Pumping Power for Various Coolants

Coolant	h, Btu/hr ft^{2}F	Pumping Power (Relative to Heat Removal Rate water)
Light and heavy water	5,000-8,000	1.0
Organic liquids (polyphenyls, etc.)	2,000-3,000	4-10
Liquid metals (Na, NaK, etc.)	4,000-10,000	3-7
Gases (He, CO_2, air, etc.)	10-100	$\sim$ 100

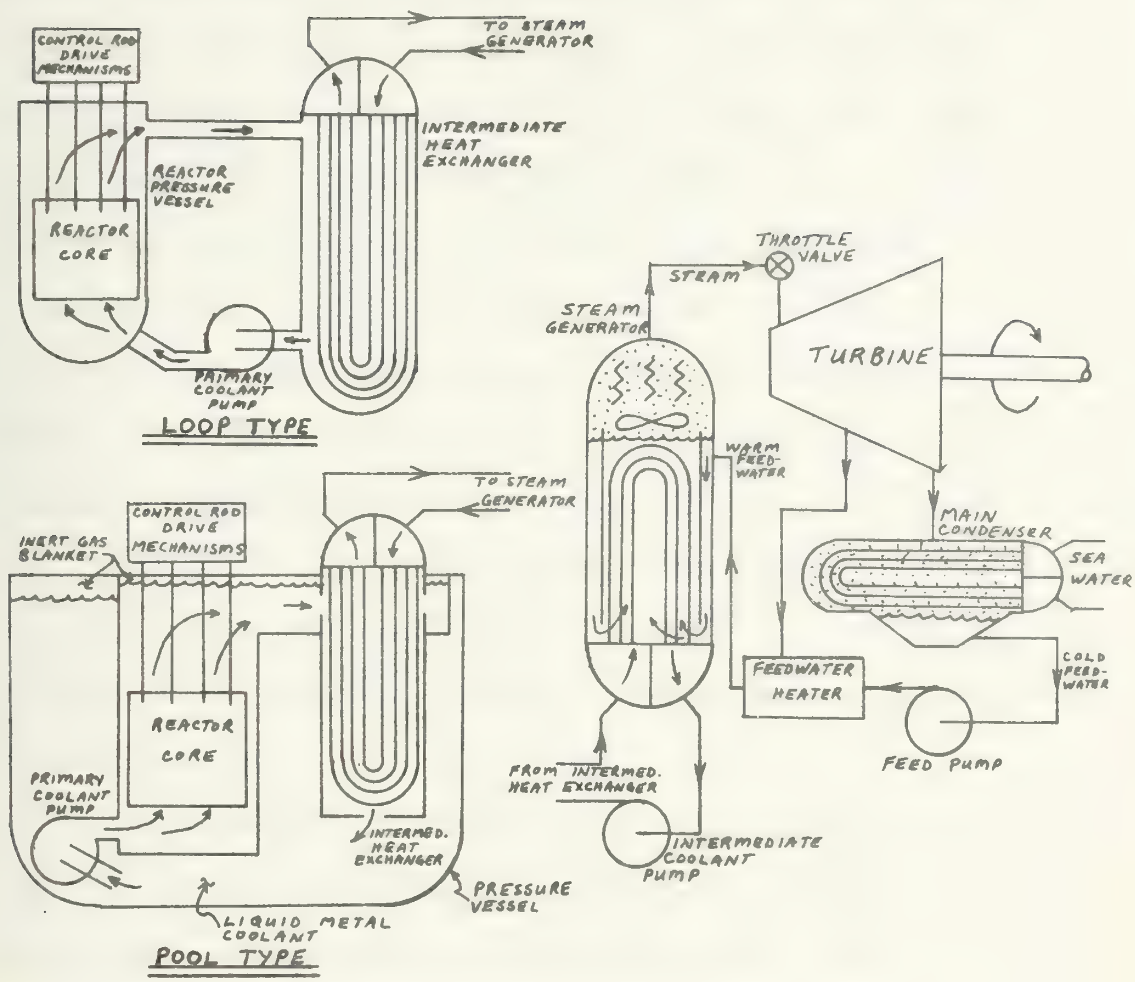

Figure II-8 Simple Flow Diagram of a Liquid Metal Cooled Reactor Power Plant

Although liquid metals are compatible with many common reactor materials such as Series 300 and 347 stainless steel and Inconel, the presence of oxygen in sodium coolants, even in minute amounts, promotes formation of Na_2O which is highly corrosive to most reactor plant materials. With a solubility that is highly temperature dependent, Na_2O also deposits in cooler regions, tending to block narrow flow passages and inhibit heat transfer in these regions. This

effect can be reduced significantly by use of a coolant purification system (such as one using a cold trap) capable of removing these corrosive oxides. The high affinity of sodium for oxygen absorption also causes reduction of oxide coatings from other plant metals in contact with each other, leading to self-welding of these metals and malfunction of such system components as pumps and valves. Further, dissolution of plant materials by liquid metals is highly temperature dependent, leading to mass transport of material from hot surfaces, where solubility is high, to cooler surfaces, where solubility is low. Although the rate of mass transfer is not great, the process is significant and must be accounted for in the design of the plant.

Because the amount of parasitic neutron capture by fertile material relative to neutron capture by fissile material in a fast reactor is greater than that in a thermal reactor, higher fuel enrichments (15% or more, compared to 1-3% in a PWR) must be used. Also, because fast-fission cross sections are a few hundred times lower than those for thermal neutrons, much more fissionable fuel is required in a fast reactor core than in a thermal core of the same power output and volume. Absence of a moderator and use of liquid metal coolants, however, allow fast reactor cores to be much more compact than thermal cores with the same power output, as can be seen from Table II-2 below.

Table II-2 Comparison of Typical Average Core Power Densities and Core Volume for a 500 MWt Reactor

Type Plant	Typical Average Power Density, MW/ft^3	Typical Core Volume, ft^3
PWR	2.5	200
BWR	0.8	650
Gas Cooled Reactor	0.25	2,000
Liquid Metal Cooled Reactor	25	20

In the usual cases, the material property limiting minimum core size is maximum permissible fuel centerline temperature in liquid metal and gas cooled reactors, and maximum permissible cladding temperatures and/or burnout in PWR and BWR plants. Burnout is localized melting of the cladding caused by steam blanketing; it is characterized by a sharp temperature rise in cladding and fuel resulting from the constant heat generation in the element and the sudden reduction of heat transfer coefficient from cladding to coolant. Low fission product cross sections for fast neutron absorption make the fast reactor less sensitive to fission product poisoning and allow higher burnups (100,000 MWD/tonne or more, compared with 30,000 for PWR and BWR).

Compared with the PWR plant, a sodium cooled plant permits approximately 10 times the temperature rise through

the reactor and several hundred degrees higher coolant temperature at the steam generators. Because of the higher steam temperature and resulting higher thermal efficiency, it can provide the same shaft horsepower with only about 85% the reactor power. These advantages do not come "free", however, since the additional shielding required tends to make the sodium-cooled plant heavier. Its capital cost also tends to be higher, and -- an important consideration for the ship operator -- it is more costly and difficult to maintain.

Nuclear safety is of somewhat greater concern with fast reactors than with thermal, since even partial core meltdown could yield a critical fuel mass, resulting in an uncontrolled nuclear excursion. Substantial leakage of hydrogenous material such as water into the core could also cause prompt critical conditions, resulting in an uncontrolled nuclear excursion due to the positive reactivity associated with extreme softening (lowering in energy) of the neutron energy spectrum in a fast reactor. This hazard would be compounded by violent chemical reaction between the liquid metal coolant and the water. The latter could be of concern, for example, if a fast reactor nuclear propelled ship were to sink with its primary coolant boundary breached.

A third nuclear safety concern is that a fast reactor designed for large power output sometimes has a relatively large positive void coefficient of reactivity. In such cases, loss of sodium coolant from the core or boiling of sodium coolant in the core could add positive reactivity by hardening

(increasing in energy) the neutron energy spectrum and decreasing the amount of nonfission neutron absorption in the fuel. (In small, low power reactors, this void effect is usually neutralized by increased leakage caused by void formation).

Some of these safety concerns can be alleviated to a certain extent by such schemes as:

1) designing the melting core to disperse into non-critical masses; for a ship, this would apply for any orientation, such as might occur during sinking;

2) using no more enrichment of the fuel than absolutely necessary; and

3) softening the neutron energy spectrum enough to give a large negative Doppler coefficient of reactivity, e.g. by using oxide fuel and/or including a small amount of moderator such as beryllium oxide.

The latter 2 schemes increase the ratio of resonance-absorbing fuel to non-resonance-absorbing fuel and increase the percentage of neutrons available for resonance absorption, thereby adding large amounts of negative reactivity due to the Doppler effect as fuel temperature increases.

Liquid metal cooled reactors have been used by several countries (e.g., U.S., U.S.S.R., U.K., France, and Germany) for land-based power production since 1951 and are currently being developed as breeder reactors. Of the 2 major types shown in Figure II-8, the pool system appears to have the edge in safety and economy and is being used by some U.S. companies and in French and British designs, while the

loop system has a more straightforward mechanical design and is being used by other U.S. companies and in Russian and German designs. The liquid metal cooled reactor plant has been applied, as an intermediate reactor, to the propulsion of 1 ship, the USS SEAWOLF, but was replaced with a PWR plant after 2 years of operation due to:

1) problems with leaks in the NaK-water boundary of the steam generator, and

2) the judgment that a liquid metal cooled plant was potentially too dangerous (e.g. a substantial fire hazard caused by even very small coolant leaks) and not nearly as reliable in casualty situations as PWR plants due to the long wait required for reactor compartment radiation levels to decay sufficiently to permit access for emergency repairs.

III. CURRENT STATUS OF NUCLEAR MARINE PROPULSION

(ref's 90,6,16 through 22,68,69,72 through 78,83,87,88,89)

This section contains the following:

1) a listing by country of known nuclear propelled ships, and

2) a survey of representative reactor types which have been either applied or designed for application to ship propulsion.

A. PRESENT SHIPBOARD REACTOR INSTALLATIONS --

The navies of the world can today muster a total of about 200 nuclear propelled ships. Although the vast majority of these ships are submarines, the total includes aircraft carriers, a cruiser, and frigates. In contrast, there exist today only 4 non-naval, nuclear propelled ships (one of these no longer in commission) owned by as many nations. An obvious conclusion one is forced to make is that the strategic and tactical advantages of nuclear propulsion outweigh the economic advantages.

1. Naval Nuclear Propelled Ships --

The strategic and tactical advantages of nuclear propulsion for naval vessels include at least the following:

a) For submarines, practically unlimited submergence time, reducing both their detectability and their vulnerability; long submergence time is made possible by the nuclear plant's ability to produce full power completely

independent of the atmosphere (note that concurrent development of long-endurance air regeneration systems was also necessary),

b) Ability to operate at or near ship design full speed for prolonged periods without a prohibitively high rate of fuel consumption; this results in capabilities such as:

1) prompt deployment to any point of need,

2) use of transit tracks that avoid bad weather or areas of increased danger of attack,

3) fulfillment of a mission, without replenishment, immediately upon completion of a high-speed transit or redeployment,

4) practically unlimited cruising range independent of vulnerable, shore-based or at-sea refueling facilities, and

5) rapid cycling in high speed transit to and from distant and less vulnerable sources of ammunition and other supplies needed to continue an engagement;
this capability is made possible by: 1) the reactor's several year core life (newest cores provide for 10 years of normal operations, so that a submarine can travel over 400,000 miles between refuelings), and 2) increased ruggedness of the propulsion plant to match the reactor plant's capabilities.

c) Availability of practically unlimited power

for operating high power demand weapons, sonars and other systems and for improving the living conditions and thereby the combat readiness and effectiveness of the personnel manning the ship,

d) For aircraft carriers, ability to project tactical air power into parts of the world far removed from politically vulnerable, land-based airfields; nuclear power, by providing increased aviation fuel and ammunition storage volume, makes the carrier relatively independent of tactically vulnerable sea-and land-based facilities for replenishment of aviation fuel, other combat consumables and provisions. Sustained high speed capability for such a ship and its escorts also enhances their ability to attack enemy shores along a greater perimeter of coastline and to evade and outrun enemy submarine attack,

e) For all surface ships, elimination of the smoke/soot problem; this reduces topside corrosion and visual detectability of the ship, facilitates landing of aircraft, and improves capability for sealing the ship against nuclear, biological and chemical attack, and

f) Release of many, vital man-hours to carry out other, more productive on-station duties; this results from elimination of frequent underway refueling and reduction of the frequency of other underway replenishment operations.

It is significant to note that all current naval nuclear propelled ships in the world derive their energy from

PWR plants. Only 1 ship, the USS SEAWOLF, was built with a liquid metal cooled (NaK) reactor plant; after 2 years of otherwise satisfactory operation, the plant was replaced in 1959 with a PWR plant due to: 1) problems with leaks in the NaK-water boundaries of the steam generators and 2) the judgment that a liquid metal cooled plant was potentially too dangerous (e.g. substantial fire hazard caused by even a very small coolant leak) and not nearly as reliable in casualty situations (e.g. due to the long wait required for reactor compartment radiation levels to decay sufficiently to permit access for emergency repairs).

Propulsion plants for naval nuclear ships differ from those for commercial ships in at least 3 important areas: 1) they must be designed and built to operate reliably and safely under conditions of combat shock; 2) they must be able to continue producing power following a partial casualty (since loss of power in an engagement could rapidly lead to loss of the ship); and 3) they must be capable of being maintained by the ship's force while underway.

A list by country of the world's known nuclear naval ships is provided below.

a. <u>United States</u> --

The U.S. Navy has in operation 93 nuclear powered submarines, 1 aircraft carrier, 1 cruiser, 2 frigates, and 1 deep submergence research vehicle. Of these 93 submarines, 41 are the ballistic missile-firing type, while the rest are

attack-type. Currently under construction are 21 more attack-type submarines, 2 more aircraft carriers, and 2 more frigates. Altogether these ships have steamed over 18 million miles; their combined 115 nuclear reactors represent an accumulation of over 800 years of operational experience. The first of these ships, the USS NAUTILUS, went to sea in 1955. Total cost of the U.S. naval nuclear propulsion program to date has been $17 billion. Table III-1 below lists pertinent details regarding these ships; all of these details are contained in reference 82. The SKIPJACK class was the first to combine nuclear power with "tear drop" shaped hull, used on all subsequent U.S. nuclear submarines; first proven by use on the conventionally powered USS ALBACORE, this hull, with its 7.8 to 1 or greater length-to-beam ratio, greatly improves underwater performance.

b. U.S.S.R.

The U.S.S.R. Navy has in operation over 95 nuclear powered submarines. Of these, at least 25 are ballistic missile-firing type, at least 37 are cruise (anti-surface ship) missile-firing type, and at least 25 are attack type. At least 15 more ballistic missile-firing type and several of the other 2 types are under construction; current Soviet production rate for nuclear submarines is 15 to 20 per year. The first of these nuclear ships went to sea in 1960. Table III-2 below lists pertinent details regarding these ships.

Table III-1 U.S. Naval Nuclear Ship Particulars

Ship Class (No. in Class)	Ship Type	Length OA/Beam/ Draft, ft	Displacement Surf/Submerged long tons	Number of Shafts	SHP	Surface/Submerged Speed, knots	Main Engines	Year Commissioned
GEORGE WASHINGTON SSBN 598 (5)	FBM submarine	381.7/33/29	5900/6700	1	15,000	20/approx 30	1 geared turbine	1959-61
ETHAN ALLEN SSBN 608 (5)	FBM submarine	410.5/33/30	6900/7900	1	15,000	20/approx 30	1 geared turbine	1961-63
LAFAYETTE SSBN 616 (31)	FBM submarine	425/33/31.5	6650/7320	1	15,000	20/approx 30	2 geared turbines	1963-67
NAUTILUS SSN 571 (1)	Attack sub	323.7/27.6/22	3530/4040	2	approx 15,000	20/20+	2 steam turbines	1954
SEAWOLF SSN 575 (1)	Attack sub	337.5/27.7/22	3720/4280	2	approx 15,000	20/20	2 steam turbines	1957
SKATE SSN 578 (4)	Attack sub	267.7/25/21	2570/2861	2	approx 6,600	20/approx 25	2 steam turbines	1957-59
TRITON SSN 586 (1)	Attack sub	447.5/37/24	5940/7780	2	approx 34,000	27/20	2 reactors 2 steam turbines	1959; 1968de-commissioned
HALIBUT SSN 587 (1)	Research sub	350/29.5/21.5	3850/5000	2	approx 6,000	15/20	2 steam turbines	1960
SKIPJACK SSN 585 (5)	Attack sub	251.7/31.5/28	3075/3500	1	approx 15,000	20/30+	2 steam turbines	1959-61
TULLIBEE SSN 597 (1)	Attack sub	273/23.3/21	2317/2640	1	2,500	15/20	turbo-electric drive	1960
PERMIT SSN 593 (13)	Attack sub	278.5/31.7/25.2	3750/4600	1	15,000	approx 20/approx 30	2 steam turbines	1962-67

Table III-1(cont'd)

Class (# in Class)	Ship Type	LOA/Beam/Draft	Displ. Surf/Sub.	# Shafts	SHP	Surf/Sub. Speed	Main Engines	Commissioned
STURGEON SSN 637 (37)	Attack sub	292.2/31.7/26	3860/4630	1	approx 15,000	approx 20/approx 30	2 steam turbines	1967-74
NARWHAL SSN 671 (1)	Attack sub	314/38/26	4450/5350	1	approx 17,000	approx 20/approx 30	2 steam turbines	1969
GLENARD P. LIPSCOMBE SSN 685 (1)	Attack sub	---	---	1	---	---/approx 25	turbo-electric drive	1974
--- SSN 688 (12)	Attack sub	360/33/32max	---/6900	---	---	---/---	2 geared turbines	1974---
NR-1	Research sub	140/12max/---	---/400	2	---	---/---	electric motors	1969
ENTERPRISE CVAN 65 (1)	Aircraft Carrier	1,123/133/35.8	89,600	4	approx 280,000	35	4 geared turbines	1961
NIMITZ CVAN 68 (2)	Aircraft Carrier	1,092/134/37	95,100	4	260,000	30+	geared turbines	1973-75
LONG BEACH CGN 9 (1)	Missile Cruiser	721.2/73.2/29	17,350	2	approx 80,000	approx 35	2 geared turbines	1961
BAINBRIDGE DLGN 25 (1)	Missile Frigate	565/57.9/29	8,580	2	approx 60,000	30+	2 geared turbines	1962
TRUXTUN DLGN 35 (1)	Missile Frigate	564/58/31	9,200	2	60,000	30+	2 geared turbines	1967
CALIFORNIA DLGN 36 (2)	Missile Frigate	596/61/---	10,150	2	---	approx 30	2 geared turbines	1972-73
--- DLGN 38 (3)	Missile Frigate	585/61/29.5	approx 10,000	2	---	30	2 geared turbines	1975-77

Table III-2 U.S.S.R. Naval Nuclear Ship Particulars

Ship Class (No. in Class)	Ship Type	Length OA/Beam/ Draft, ft	Displacement Surf/Submerged long tons	Number of Shafts	SHP	Surface/Submerged Speed, knots	Main Engines	Year Commissioned Or Reported
"N" Class (13)	Anti-submarine	360.9/32.1/24.3	22,500	2	22,500	20/25-30	steam turbines	1960
"E-I" Class (4)	Cruise Missile Submarine	383.9/33/27	4600/5000	---	22,500	20 max	steam turbines	1961
"E-II" Class (27)	Cruise Missile Submarine	393.7/33/27	5000/5600	---	22,500	20 max	steam turbines	1963
"C" Class (5)	Cruise Missile Submarine	385.9/32.8/24.6	4000/5000	---	24,000	---/approx 30	steam turbines	1969
"V" Class (7)	Anti-submarine	285.4/32.8/26.2	3600/4200	---	24,000	26/30+	steam turbines	1969
"H-II" Class (9)	Ballistic Missile Sub	344.5/33/25	3700/4100	---	22,500	20	steam turbines	---
"Y" Class (33)	Ballistic	426.5/34.8/32.8	8000/9000	---	24,000	22	steam turbines	1969

c. United Kingdom --

The British Navy has in operation 10 nuclear powered submarines, including 6 attack type and 4 ballistic missile type. The reactor plants in these submarines are based on technology and design details developed by the U.S. for its nuclear submarines. Table III-3 below lists pertinent details regarding these ships. All of these submarines have "tear drop" shaped hulls for maximum underwater efficiency.

d. France --

The French Navy has in operation 2 nuclear submarines of the ballistic missile type and is building 2 more. The reactor plants in these submarines, as in the British nuclear submarines, are based on technology and design details developed by the U.S. for its nuclear submarines. Table III-4 below lists pertinent details regarding these ships.

2. Commercial Nuclear Propelled Ships --

The 4 nuclear propelled non-naval ships existing today are as follows; these ships and their propulsion plants are described briefly below in the following order, and are described in more detail in Appendix I:

1) United States -- N.S. SAVANNAH (decommissioned)

2) W. Germany -- N.S. OTTO HAHN

3) U.S.S.R. -- N.S. LENIN

4) Japan -- N.S. MUTSU

Table III-3 United Kingdom Naval Nuclear Ship Particulars

Ship Class (No. in Class)	Ship Type	Length OA/Beam/ Draft, ft	Displacement Surf/Submerged long tons	Number of Shafts	SHP	Surface/Submerged Speed, knots	Main Engines	Year Commissioned
DREADNAUGHT S 101 (1)	Prototype Attack Sub	265.8/32.2/26	3500/4000	1	SKIPJACK plant	approx 30	geared steam turbines	1963
VALIANT S 102 (5)	Attack Sub	285/33.2/27	3500/4500	1	basically SKIPJACK plant	approx 30	geared steam turbines	1966--- (5 in comm.)
SWIFTSURE S 107 (4)	Attack Sub							
RESOLUTION S 22 (4)	Ballistic Missile Sub	360/33/30	7500/8400	1	"English" plant	20/25	geared steam turbines	1967-69

Table III-4 French Naval Nuclear Ship Particulars

Ship Class (No. in Class)	Ship Type	Length OA/Beam/ Draft, ft	Displacement Surf/Submerged long tons	Number of Shafts	SHP	Surface/Submerged Speed, knots	Main Engines	Year Commissioned
LE REDOUTABLE SNLE 1 (4)	Ballistic Missile Sub	420/34.8/32.8	7500/9000	1	15,000	20/25	2 turbo-alternators; 1 electric motor	1971-76 (2 in comm.)

In addition to these ships, the following are reportedly under construction (ref's 82, 91, 92, 93), with the current status not known:

1) Italy -- The Italian Navy and the CNEN (National Committee for Atomic Energy) signed a cooperative agreement in 1968 to design and build a 9,277 dwt, 18,000 ton displacement, 22,000 SHP, 574 ft long, 20 knot nuclear propelled ship. This ship, tentatively named ENRICO FERMI, is intended to be operated by the Italian Navy as a logistical support ship which will train civilian crews in PWR reactor plant operation. (ref's 29 and 91)

2) U.S.S.R. -- The success of LENIN is attested by the fact that the U.S.S.R. is currently completing construction of 2 more nuclear icebreakers and is planning a fourth (80,000SHP) to be built by the Wärtsilä yard. The 2 icebreakers under construction are of the ARKTIKA class and have the following major characteristics: (ref 82)

Displacement	25,000 tons
Length	524.9 ft
Beam	82.0 ft
Depth	33.5 ft
Reactors	2, PWR
Shaft horsepower	30,000
Speed, open water	25 knots
Helicopters carried	10

3) China -- With Russian technical assistance, the Chinese Nuclear Energy Commission and the Ministry of Transport are reported to have developed and to be building the ZAN THAN ("Voice of the People"), a 22,000 SHP each, 2 screw, 22,000 gross ton, 23.5 knot ferry with a 180 MWt PWR propulsion plant, and the BAC PHAN, a 50-60,000 SHP ferry with a 210 MWt nuclear propulsion plant. (ref's 92, 103, 104) Moreover, Japan and West Germany have recently announced an agreement to mutually build two 80,000 SHP, high speed, nuclear powered container ships as a start for their nuclear merchant fleet. These 51,000 gross tonnage ships will carry 1,840 containers at a service speed of 26 knots. (ref's 92, 93)

N.S. SAVANNAH -- (ref's 23, 24, 26, 40 through 44)

Authorized by Congress in Public Law #848 in 1956, N.S. SAVANNAH was intended to further demonstrate to the world the United States' sincere interest in the peaceful uses of atomic energy for the improvement of human living. Named after the first steam-powered vessel to cross the Atlantic, the world's first nuclear merchant ship was designed by the Babcock and Wilcox Co. (reactor plant) and George G. Sharp Inc. (ship) and built by the New York Shipbuilding Corp.; propulsion equipment was supplied by the DeLaval Steam Turbine Co. Government administrative control was provided by two newly established groups: in the A.E.C., the Maritime Reactors Branch of the Division of Reactor Development; and in the Dept. of Commerce, the Nuclear Projects Office of the Maritime Administration (MarAd). SAVANNAH construction was begun in May, 1958; the ship was launched in July, 1959 and initial reactor criticality was achieved on Dec. 22, 1961. SAVANNAH was delivered May 1, 1962 for final testing, sea trials, and commercial operation by States Marine Lines of Delaware.

SAVANNAH is a modified "Mariner" type hull design with a maximum speed of 21 knots at a propulsion power of 22,000 SHP. With accommodations for 60 passengers and general cargo of 10,000 tons, her length is 595 ft 6 in., beam is 78 ft, and draft is 29 ft 6 in. Other major characteristics of SAVANNAH are as follows; additional details are provided in Appendix I:

Displacement, light ship	11,850 long tons
Plant Efficiency	23.2%
Reactor	1 pressurized light water, 70 MWt maximum
Fuel System	Sintered Uranium Dioxide pellets in stainless steel tubes; 4.4% average enrichment
Machinery weights, total:	3,890 long tons
Nuclear plant, including radiation shielding	2,760 long tons
Machinery plant other than nuclear	1,130 long tons
Steam drum steam conditions	472 psia, 460F, saturated, quality $\geq$ 99.75%

Some pertinent facts about the first steam-powered SAVANNAH are as follows: Powered by a single cylinder, 90 HP, reciprocating steam engine with a 40 in. bore and a 5 ft stroke, this 320 ton ship cost $50,000, almost twice the cost of a comparable conventional sailing vessel in those days. A monumental first in transoceanic transportation history, she was commercially a colossal failure. Since her engine required nearly a ton of coal/wood for each hour of operation, it could be run for only 80 hours of the 29 1/2 day Atlantic crossing from Savannah, Ga. to Liverpool. The construction of her amidships-mounted paddle wheels was unique; their 8 radial arms and paddle blades were arranged to fold like a

fan for non-use storage in an upraised position for protection from heavy seas. Like her later counterpart, the earlier SAVANNAH also had manning problems; the "steam coffin" was deemed so hazardous that only the fine reputation of her officers in their own home town induced the crew to ship aboard the craft. The successful return trip home found her owners so dissatisfied with her high operating cost that her engine and paddle wheels were removed and she was put in the coastal cotton trade. The stormy history of the SAVANNAH that ushered in the age of the steam-powered merchantman was later to be matched by that of the SAVANNAH that ushered in the age of the nuclear-powered merchantman.

The mission of the nuclear SAVANNAH was threefold:

1) Serve as a prototype to test the adequacy of design criteria developed for nuclear-powered merchantmen and test the adequacy of components developed for use in merchant ship nuclear propulsion plants.

2) Train merchant marine personnel in nuclear plant operation and maintenance, and develop satisfactory operating procedures for these plants, and

3) Ensure acceptance of nuclear-powered merchantmen in all ports of the world by stimulating early solutions to such problems as international liability and indemnification and by public demonstration of the safety and dependability of nuclear propulsion. To this end, SAVANNAH was consciously designed to be a general purpose passenger/ cargo vessel capable of sailing many different trade routes,

rather than a more economical ship such as a tanker designed for 1 specific route carrying 1 type cargo in great quantities.

SAVANNAH's first domestic demonstration voyage commenced in late August, 1962 and ended in February, 1963. A prolonged labor dispute deactivated the vessel from May, 1963 to May, 1964 and resulted in contracting of a new general agent, American Export Isbrandtsen Lines (AEIL), to continue planned demonstration tours of domestic and foreign ports. In August, 1965 a Bareboat Charter Agreement was executed with First Atomic Ship Transport (FAST), a wholly owned subsidiary of AEIL, for experimental commercial operation of the ship. This agreement was extended to include the first refueling, involving replacement of 4 of the original 32 fuel elements, and continued operation through FY 1969. This first refuelling began August 23, 1968 and was completed in 2 months time; an increase in operating time of 8000 effective full power hours was thereby added to the core life over and above the 15,500 EFPH expended prior to refueling.

In late 1968 MarAd solicited proposals on the part of any competent U.S. flag steamship operator for a Bareboat Charter Agreement or for outright transfer of title. In spite of the prospect of zero fuel costs for 5-6 years due to availability of a pre-paid replacement core with 4 additional spare elements for use when the installed core was expended, no acceptable proposals were received. The ship was removed from commercial service in July, 1970 and decommissioned at the nuclear servicing facility of Todd

Shipyards in Galveston, Texas. Her core removed and her power plant fluid systems drained, SAVANNAH is scheduled to be transferred to her namesake port in Georgia in January, 1972 for final deactivation. SAVANNAH logged 450,000 sea miles and called on 26 countries, winning public confidence and acceptance, yet was precluded by her very design from operating as an economically viable member of the American merchant fleet.

N.S. OTTO HAHN -- (ref's. 28 through 32,34,35,36,43,44,66,67,71)

Appropriately named after a pioneer in the discovery of the nuclear fission process from which she derives her power, the N.S. OTTO HAHN is a 25,900 ton, 14,000 deadweight ton, nuclear research ship/ore carrier. The ship was built at a cost of $14,000,000 (plus fuel) by Kieler Howaldswerke shipyard in Hamburg for Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt (GKSS), a German public utility company formed in 1956 to promote application of nuclear power for merchant shipping. Her 10,000 SHP nuclear power plant, supplied by the Deutsche Babcock-Internationale Atom-reaktorbau GmbH Consortium (Interatom) is a slightly modified version of the Babcock and Wilcox-designed Consolidated Nuclear Steam Generator (CNSG-I).

This plant, licensed to Germany for OTTO HAHN, embodies the concept of an integral reactor (steam generator within the reactor pressure vessel) and is a product of considerable United States design effort toward upgrading and improving the N.S. SAVANNAH reactor plant design to produce a lighter, more efficient, more easily automated and more economical plant for use in subsequent nuclear merchantmen. This plant, redesignated the FDR (Fortschrittlicher Druckwasserreaktor) for its application in OTTO HAHN, was selected from 4 proposed designs: The Organic Moderated Ship Reactor (OMSR); the Gas Cooled Ship Reactor (630S); the Siemens Pressurized Water Reactor; and the CNSG-I.

Half the $7,000,000 capital cost for the 38MW pressurized light water reactor plant was funded by Euratom, the Common Market's AEC, in return for operating data. The remainder of the ship's cost was funded by the German Bund and by the 4 Northern German coastal districts, Bremen, Hamburg, Niedersachsen and Schleswigholstein.

Initial GKSS design studies, begun in 1958 and centering around a conventional tanker converted to nuclear power, indicated that a bulk carrier would best suit the ship's primary purpose of nuclear research, since its ample tank capacity for water ballast would favor full-draft, full-power trials independent of cargo availability. Other desirable advantages of the bulk carrier over a tanker were that the ship can be run for different trades on different routes to various ports, thus enabling the owner to gain the widest experience; it can be more readily designed to withstand collisions and groundings; the danger of fire is lower; and the navigational and technical experiences of a bulk carrier would be of interest to a larger number of ship owners. Economic profitability of the ship was considered subordinate to other design criteria which would improve its usefulness as a nuclear research ship.

The shipbuilding contract was signed November 28, 1962, the ship's keel was laid September 17, 1963, and launching took place June 13, 1964. The OTTO HAHN reached initial criticality August 26, 1968 and completed sea trials October 11, 1968. Since then she has sailed on many research voyages under

different sea conditions and in different areas, with high reliability of both ship and reactor plant; her design full load speed is 15 3/4 knots. Other major characteristics of OTTO HAHN are as follows; additional details are provided in Appendix I:

Plant Efficiency	20.8%
Reactor	1 pressurized light water, 38 MWt maximum power
Fuel System	Sintered Uranium Dioxide pellets in stainless steel tubes; average enrichment 4.02%
Machinery weights, total:	3,050 tons*
Nuclear plant, including radiation shielding	2,050 tons
Machinery plant other than nuclear	1,000 tons
Steam drum steam conditions	456 psig, 523.4F, 65F superheated

* The 1,000 ton weight of the reactor service room and the machinery and equipment contained therein is not included in these weights.

N.S. LENIN -- (ref's 18, 44, 55 through 59)

The LENIN is a 16,000 ton, turboelectric-drive icebreaker powered by 3 pressurized light water reactors, any two of which can supply full power. Built at the Kirov Elektrosia Works in Leningrad by the U.S.S.R., LENIN is the world's first non-naval, nuclear-powered ship; she joined the Arctic fleet December 3, 1959, and has aided the advancement of the economic development of the Soviet Northern Regions since that date. The use of nuclear power on an icebreaker provides greatly improved operational capabilities; among these are the following: 1) greater open water speed and greater icebreaking capability, both due to greater installed power, 2) more prolonged periods on station due to freedom from frequent bunkering, and 3) no lack of auxiliary power should the vessel become icebound.

Because the LENIN was Russia's first operating nuclear propulsion plant and because its service entailed long durations away from base support, a high degree of reliability and maintainability was designed into her propulsion plant. The provision of 3 reactors, with double coolant loops for each, plus double armature electrical generators and propulsion motors, multiple sources of power and 100% redundancy of pumps all ensure availability of equipment for routine, preventive maintenance and only slight if any reduction in capability due to failure of any one component.

LENIN's other major characteristics are as follows; additional details are provided in Appendix I:

Shaft horsepower	44,000 split between the 3 shafts in the ratio 1:2:1
Plant efficiency	18.4%
Reactors	3, pressurized light water; 90 MWt maximum each
Fuel system	Sintered Uranium Dioxide pellets in zircalloy tubes; average enrichment 5%
Steam drum steam conditions	412 psia, 590F, 142.5F superheat
Machinery weight, total	5,767 tons
Nuclear plant, including radiation shielding	3,017 tons
Machinery plant other than nuclear	2,750 tons

N.S. MUTSU -- (ref's 44 through 54)

As early as 1956 Japan showed interest as a maritime country in nuclear merchant ship propulsion. In the early 1960's the Japan Nuclear Powered Ship Research Association, a private organization established in 1958, completed a conceptual design for a nuclear powered oceanographic survey ship. In 1963 The Japan Nuclear Ship Development Agency (JNSDA) was established as a public corporation to effect the construction of this ship. Budget limitations dictating the necessity of at least partial recovery of the ship's capital investment, its design was soon changed to that of a special freighter to be used for transporting nuclear fuel; the scarcity of uranium reserves in Japan and the aggressive transition of that country's main source of energy to nuclear power virtually guarantees a continuing demand for the ship's services. In addition, it will serve as a training ship for nuclear plant operators and will have provided Japan with significant experience in the design, construction and operation of nuclear ships; this experience should enable Japan to enhance her present position as a leading country in the shipbuilding and maritime industry.

The nuclear powered special freighter, MUTSU, is a 10,000 SHP, 16.5 knot, single screw vessel of 8,350 gross tons and 2,400 deadweight tons. Her single, 36 MWt pressurized light water reactor plant is similar in basic configuration to that in the N.S. SAVANNAH. Construction of the ship was begun by Ishikawajima-Harima Heavy Industries Co. Ltd. of

Tokyo, November 27, 1968, under an $8M contract which included the ship's hull, turbine, electrical system, reactor containment vessel, and secondary shielding; the ship was launched June 12, 1969. With the conventional portion of the ship completed, she sailed under auxiliary boiler power to her base harbor and namesake city Mutsu, where her reactor plant was installed by Mitsubishi Atomic Power Industries, Inc. under a $7.4M contract. Completion of the ship is scheduled for early 1972.

Under the administrative and engineering control of the JNSDA, which will act as owner-operator of both the ship and its support facilities in Mutsu Harbor, the ship was designed and constructed using established and specially developed Japanese domestic technology, equipment, and materials to the maximum extent possible; only a very limited number of materials, such as uranium, was imported. Three-fourths of the estimated $33M cost for development and construction of the ship and its land-based support facilities at Mutsu has been funded by the Japanese government, the remainder by interested private companies. Mutsu was chosen for a home port after other prospective ports, including Yokohama, declined on the basis of sea traffic congestion or fear of nuclear contamination. Other major characteristics of MUTSU are as follows; additional details are provided in Appendix I:

Plant efficiency	23%
Fuel System	Sintered UO_2 pellets in stainless steel tubes; radial enrichment zones of 3.2 w/o and 4.4 w/o U-235.
Machinery weights, total	tons
Nuclear plant, including radiation shielding	tons
Machinery plant other than nuclear	tons
Steam drum steam conditions	568 psig, 484F, saturated

00073c

B. EXISTING REACTOR TYPES AND DESIGNS FOR MARINE APPLICATION --

Section II (Background) presents details of the major types of reactor plants which are currently suitable for large scale power production. In evaluating the suitability of these and other reactor plant types for specific ship propulsion application, many factors must be considered, such as: nuclear safety in the marine environment; propulsion plant weight and volume; plant capital, operating and maintenance costs; expected plant reliability under conditions of shipboard vibrations and ocean-induced forces; and degree of risk involved due to such variables as extent of use of unproven or developmental technology or equipment.

Section III. A. above presented details of the reactor plants that have been used to date for ship propulsion. Construction and operational experience with these ships has emphasized the desirability of designing future nuclear propulsion plants with certain improved characteristics, such as the following:

1) lighter and smaller, so as to consume less of the ship's weight and space; these features would result either in a smaller, lighter, faster ship or in more space available for payload such as cargo or weapons.

2) cheaper and faster construction, and cheaper operation; these features would give additional incentive for construction of nuclear ships by 1) requiring less tying up of shipowner capital and shipbuilder facilities and manpower, and 2) providing a larger return on the resources invested in the ship.

Many conceptual studies and several complete, detailed designs for improved nuclear propulsion plants have been made, both by government agencies and private industry, in an attempt to incorporate improved characteristics such as the above. Some of the features incorporated in these designs include:

1) Consolidation of one or more of the primary coolant loop components inside the reactor pressure vessel;

this can reduce both plant volume and weight since the increase in size and weight of the reactor vessel can be more than offset by: a) the decrease in volumes and weights of the reactor containment vessel, the secondary shield, and the collision barrier, and b) elimination of some or all of the heavy-walled primary coolant boundary, such as the piping itself and the high pressure walls of the steam generators and the pressurizer.

2) Prefabricability of part or all of the reactor plant in vendor or shipyard shops physically removed from the shipbuilding ways, and rapid, modular installation of these parts into the ship; this eliminates from the construction yard the costly and time-consuming operations associated with assembly, welding and flushing of high pressure fluid systems, and decreases the total time and cost required for construction of the ship.

3) Longer life core that can be rapidly replaced as a module; this allows the ship to operate for longer periods of time between refuelings and reduces the outage time required for each refueling.

4) Simplification of principal reactor systems by combining functions and eliminating redundant components.

5) Improvement of plant thermal efficiency by provision of higher temperature, superheated steam at pressures corresponding to those used in conventionally heated steam plants.

Pertinent details of some of the more promising reactor plants designed specifically for marine propulsion are presented below. A more detailed description of each of the first 3 plants is included in Appendix I. Additional details of the other plants can be found in the references cited. It should be noted here that this survey does not include in detail the many designs, undertaken in the late 1950's and early 1960's by many different organizations in several countries, which were based on then-undeveloped or unproven technology. An example of such a design is that for a gas cooled reactor plant with direct, closed cycle gas turbine. This plant has recently received much development work for central station application, especially in Switzerland, so that with today's technology it appears feasible to obtain outputs in excess of 300 MWe with thermal efficiencies in excess of 40% from this plant. Prior to the recent development of adequate heat-resistant materials, reliable, leak tight seals for turbine and gas circulator shafts, and efficient, high power gas turbines, this plant could not utilize optimum pressures needed for compact cores or the gas temperatures well in excess of 1250F needed to obtain high thermal efficiencies (below this temperature the efficiency of a gas turbine cycle drops off sharply and is less than that of the steam turbine cycle).

1. <u>The Babcock and Wilcox Consolidated Nuclear Steam Generator (CNSG)</u> --

The CNSG is a light water cooled and moderated

PWR plant, suitable for SHP's in the range 20,000 to 160,000, which substitutes reactor vessel internal flow passages for the bulky primary coolant piping; the 120,000 SHP version is described below. The CNSG's 4 steam generators and 4 primary coolant pumps are located inside the 30.5 ft high, 13 ft ID, 6.5 in. thick reactor pressure vessel. A separate pressurizer is provided to maintain 1850 ± 25 psia primary coolant pressure. The 34 ft diameter, 48 ft high containment vessel is surrounded by a 2 ft thick aggregate concrete secondary shield. The CNSG plant is smaller, but heavier, than the boilers it can take the place of. Compared to the SAVANNAH reactor plant, the 70,000 SHP CNSG design produces 2 1/2 times the power, in half the volume, with half the weight.

Primary coolant core inlet/outlet temperatures are 572.3F/604.5F. Steam generator steam outlet conditions are 700 psia, 553F, 50F superheated steam with a steam flow rate of 1,224,000 lb/hr. Feedwater temperature is 400F and condenser pressure is 2 in. Hg. The 312.5 MWt core is that of a central station PWR with additional fuel rod lateral support for vibration and seaway motion; the 8,640 fuel rods are 85.75 in. high, 0.430 in. OD, 0.0265 in. wall thickness zircalloy-4 tubes containing 0.370 in. OD, UO_2 pellets with an average U-235 enrichment of 4.7%. Fuel burnup at batch type refueling is 35,000 MWD/t, equivalent to a 5 year core

life at 70% load factor. Additional details are presented in Appendix I.

2. The General Electric 630A Maritime Nuclear Steam Generator --

Basically an extension of the technology developed for the Aircraft Nuclear Propulsion Program between 1951 and 1961, the 630A is a helium-cooled, water-moderated reactor plant producing steam for use in a conventional geared-turbine propulsion plant. The design emphasizes modular fabrication and factory preassembly; rapid shipboard assembly is facilitated by factory-prepared cable runs and use of pin-type electrical connectors. The 630A is suitable for SHP's in the range 2,000 to 120,000; the 27,300 SHP version is described below. Consolidation of primary components is achieved by placement of the 2 steam generators inside, and the 2 gas circulators in domes flanged to, the 9.5 ft ID, 24 ft high reactor pressure vessel. The 12.5 ft diameter, 39 ft high reactor containment vessel is surrounded by a sheet of lead and encircled in the region of the core by a 26.5 in. thick tank of borated water for shielding. The 630A plant is about the same size as, but somewhat heavier than, the boiler(s) it can replace.

The 825 psia helium primary coolant core inlet/outlet temperatures are 553F/1200F. Steam generator steam drum conditions are 1535 psia, 1005F, 405 F superheat with a steam flow rate of 172,800 lbs/hr. Feedwater temperature

is 415F and condenser vacuum is 1.5 in. Hg. The 60.5 MWt core can be either of 2 designs, containing either 109 or 151 fuel cartridges 42 in. high, each cartridge holding either 55 or 38 fuel rods. Fuel rods are either 0.350 or 0.390 in. OD, both 0.015 in. wall thickness incoloy tubes filled with 5 w/o U-235 enriched UO_2. The feedwater-cooled moderator flows through the core in helium-tight channels and provides core top, bottom and side reflectors; one of the 2 core designs also employs a beryllium side reflector. Fuel burnup is 15,000 MWD/t, or 17,350 effective full power hours; refueling for both core designs is by modular replacement of the core/moderator housing complex. Additional details are presented in Appendix I.

3. The Combustion Engineering Unified Modular Plant (UNIMOD) --

Emphasizing factory preassembly into a small number of modules suited for rail shipment and rapid, simple shipboard installation, the UNIMOD plant is a light water cooled and moderated PWR plant suitable for SHP's in the range 10,000 to 60,000; the 30,000 SHP version is described below. As in the OTTO HAHN plant, moderate boiling of coolant in the core maintains a steam dome at the top of the reactor vessel, providing adequate primary coolant pressure regulation (at saturation pressure for core outlet temperature) without use of a separate pressurizer. The 6 steam generators are mounted inside, while the 3 primary coolant pumps are outside, the 71 in. ID, 22 ft 4 in. high reactor pressure vessel. The

pumps and the vessel are connected by 3 short runs of concentric primary coolant piping. The 16 ft diameter, 34 ft high reactor containment vessel has no external shielding. The required shielding is provided by concentric iron rings and lead slabs in an annulus of borated water; this water fills the space between the reactor pressure vessel and the containment vessel to a level above the reactor vessel head. The containment vessel together with its entire contents weighs 430 tons.

Primary coolant core inlet/outlet temperatures are 610/652 F, giving steam drum conditions of 600 psig, 600F, 112F superheat with a steam flow rate of 300,000 lb/hr. The 80MWt, 2-pass core contains 61 fuel assemblies each of which contains 126, 0.328 in. OD, 0.015 in. wall stainless steel tubes filled with an average 5.9 w/o U-235 enriched UO_2. A unique feature of the UNIMOD plant is that the control rods, which are mounted on top of moveable fuel assemblies, are fully withdrawn throughout all ranges of reactor operation; reactivity is controlled by the inherent core features such as Doppler, temperature and void coefficients. Fuel burnup is 20,000 MWD/t, giving a core life of 3.4 years at 80% load factor. Additional details are presented in Appendix I.

4. The NERO Nuclear Ship Propulsion Plant -- (ref 84)

Developed by the Netherlands Reactor Centre from 1961 to 1967, the NERO plant was designed as a reactor

system that would be as simplified and reliable as possible. Economic competitiveness was to be assured by designing for minimum fuel cost; this requires maximum fuel burnup, simple fuel element design, and fairly large power density. Most of the development effort for this plant was funded by Euratom.

The NERO is a light water cooled and moderated PWR plant suitable for power levels up to 120,000 SHP; the 22,000 SHP design is described here. As shown in Figures III-1 and III-2, the primary coolant system consists of 2 parallel piping loops connected to a 78.8 in. ID, 19 ft 1 in. inside height reactor vessel. Each of these 2 loops contains a vertical U-tube steam generator, a horizontal U-tube steam superheater, a 260 kw, 380 v, 3 phase, 4 pole, canned induction motor coolant pump, 2 remote-operated loop isolation valves, and a check valve to prevent reverse flow.

An electrically heated pressurizer maintains primary pressure at 2130 psia. Heat transfer surface areas of the steam generator/superheater are 1835 ft^2/598 ft^2. Use of the superheater results in several advantages:

a) no moisture separators are needed upstream of the turbine throttle valve,

b) the problem of condensate flashing in the steam lines during power maneuvers is eliminated,

c) throttle valve erosion at low power levels is greatly reduced,

d) cycle thermal efficiency increases several

percent, and

e) turbine operating life is increased.

The usual auxiliary systems required for reactor operation and maintenance are provided. All radioactive fluids are contained inside the 29 ft 7 in. inside diameter spherical containment vessel. Secondary shielding consisting of lead, concrete and polyethylene is situated both inside and outside the containment vessel. The weight of the entire reactor system, including shielding and containment vessel, is 1,060 tons.

The 63 MWt core consists of 12 identical, hexagonal fuel elements. Each fuel element contains 282, 0.429 in. OD, 0.0343 in. wall thickness zircaloy-4 tubes packed with 0.394 in. OD, dished, 6 w/o uniformly enriched UO_2 fuel pellets. UB_4 burnable poison is incorporated in these fuel pellets to achieve long core life without fuel shuffling or excessive control rod movement; two radial zones concentration of this poison are used to achieve power distribution flattening and higher average power density. Each fuel element also contains 48 guiding thimbles for the sintered B_4C control rods. The 48 control rods for each fuel element are connected to a common shaft above the core and driven by a single control rod drive mechanism of the rack and pinion type. Primary coolant core inlet/outlet temperatures are 554F/571.4F. Full power steam flow is 224,000 lbs/hour; steam conditions at the outlet of the superheater are 582 psia, 545F, 62F superheated; feedwater temperature is 410F.

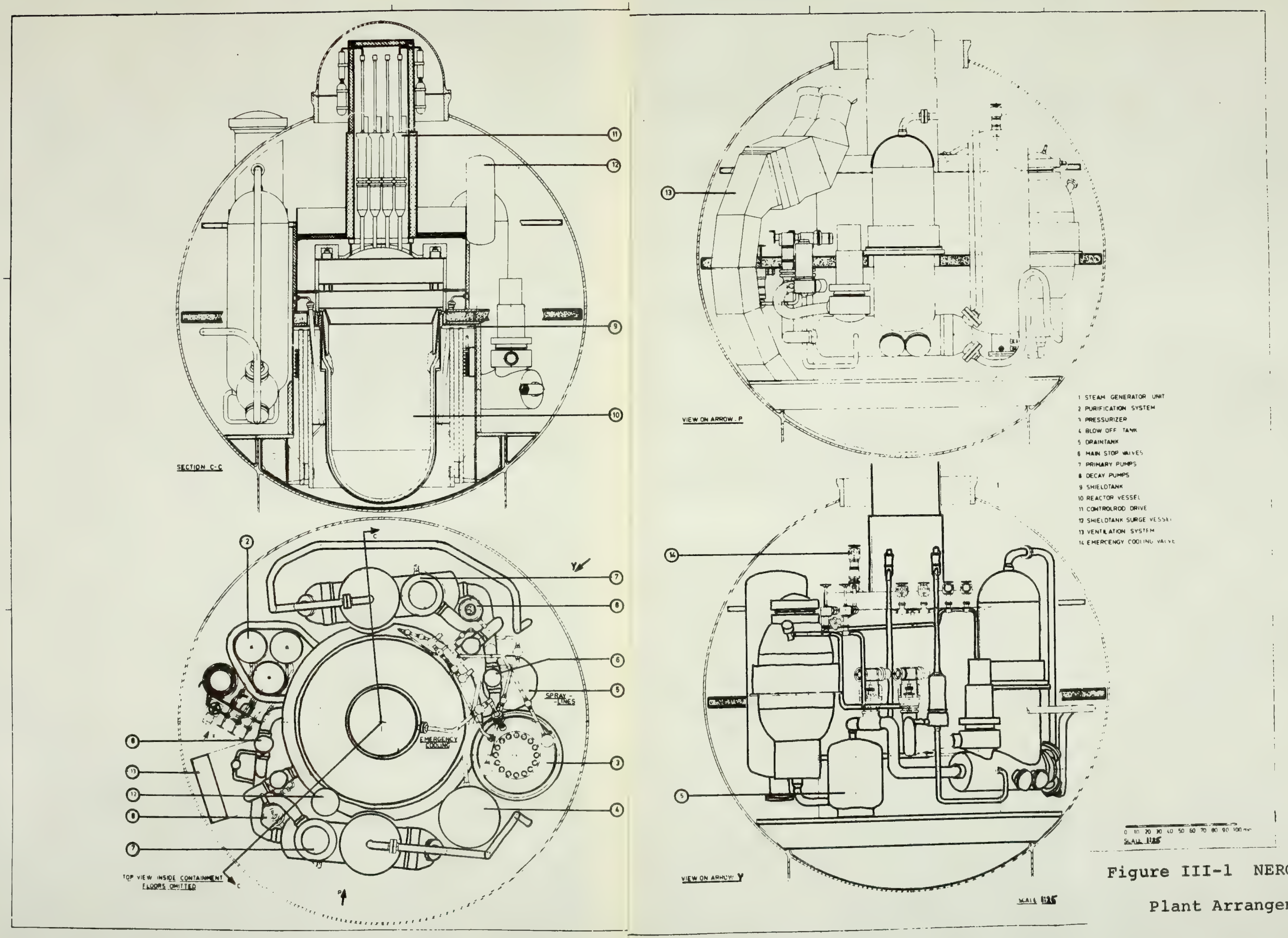

Figure III-1 NERO Reactor Plant Arrangement

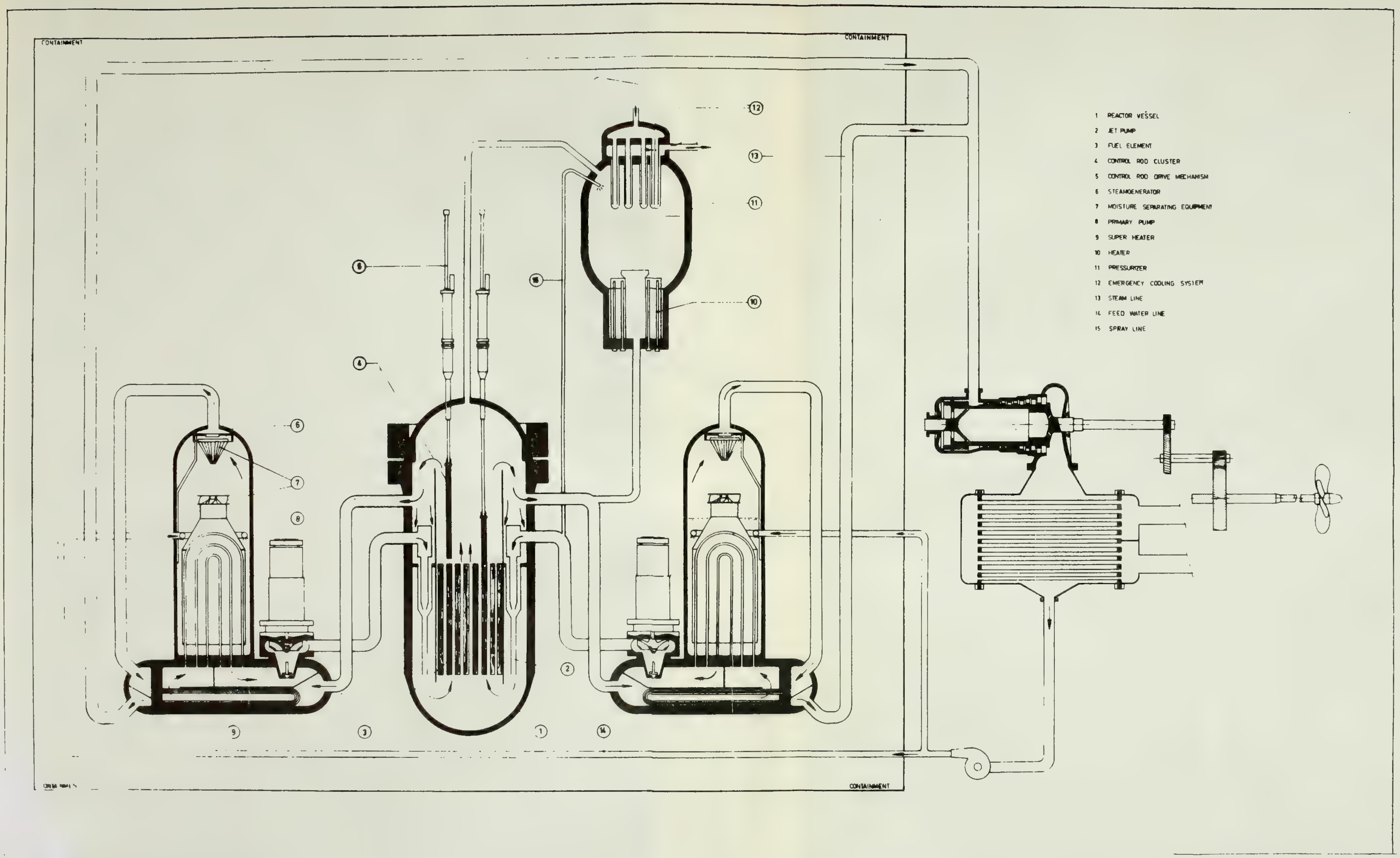

Figure III-2 NERO

Reactor Plant Simplified

Schematic Diagram

Core life is designed to be 26,400 equivalent full power hours, approximately 4 calendar years of ship operation; refueling is batchwise to minimize refueling time. Primary flow through the core is single pass with internal recirculation (recirculation ratio 2.59) within the reactor vessel provided by 30 jet pumps located peripherally between the core and the vessel wall (see Figure III-2). Throat diameter in these pumps is 2.52 in. The use of these jet pumps reduces external flow rate so that smaller, more compact loops can be used, and provides sufficient natural circulation of the coolant to remove core decay heat to the upper plenum of the vessel in the event pumping power is lost. Natural circulation flow between this plenum and the pressurizer then transfers the decay heat to the pressurizer. A third natural circulation cooling loop (not using primary coolant) between the pressurizer and an air cooled condenser on the upper deck then transfers the decay heat to the atmosphere. A fully automatic control system sensing steam pressure and coolant temperatures moves control rods to maintain constant coolant core outlet temperature, permitting faster maneuvering rates with smaller pressurizer in-and out-surges.

5. <u>The Westinghouse Nuclear Propulsion Plant for High-Speed Merchant Ships</u> -- (ref 86)

Based entirely on central station proven design concepts and component technology, the Westinghouse high-speed merchant ship reactor is a light water cooled and moderated PWR of the loop type. Designs for this plant have been

completed for SHP's up to 140,000; the 75,000 SHP version will be described here. Westinghouse's strong selling point for this plant is high reliability and sustained operation with little maintenance; in summary, minimum financial risk for the owner/operator. The containment vessel is a vertical cylindrical, steel and concrete structure which is an integral part of the ship's structure. With an OD of 31 ft, a height of 37 ft to the top of its ellipsoidal head, and a 33 in. thick concrete wall, the containment vessel is situated at the forward end of the 83 ft wide, 90 ft long, 37 ft high engine room. The weight of the reactor plant, including containment vessel and shielding, is 1,636 tons; the entire propulsion plant weights 3,023 tons.

The primary system is arranged in the containment vessel to provide the most compact plant consistent with good maintenance accessibility to all components. The 8 ft 10 in. ID, 23 ft high cylindrical reactor vessel with ellipsoidal bottom and bolted hemispherical top is located at the forward edge of the containment vessel. The 2 primary coolant piping loops are arranged with transverse symmetry in the containment vessel. Each 15 in. ID stainless steel piping loop contains: a vertical, Inconel U-tube steam generator with integral moisture separator; 23,000 gpm, vertical, single stage, radial flow, 2 speed, canned ac induction motor coolant pump; and 2, motor-operated, gate type loop isolation valves. Coolant core inlet temperature is 550F and outlet temperature

is 583F; flow rate through the core is 46,000 gpm. Coolant pressure is maintained at 2,000 psia by a 4 ft 2 in. OD, 29 ft high cylindrical pressurizer with replaceable electric heaters of 280 kw total capacity. Steam generator full power steam drum conditions are 620 psia, 490F saturated steam with $\geq$ 99.75% quality, at a total steam flow rate of 810,000 lb/hr.

The 220 MWt core has a 6 ft active height and a 6 ft 6.5 in. effective diameter. The core consists of 52, "canless" (open-sided) fuel assemblies. Each of these assemblies is a 13 x 13, 0.741 in. pitch, square cross sectional array of 149, 0.552 in. OD, 0.030 in. wall thickness zircaloy tubes containing: 1) 0.486 in. diameter, 3.7 w/o U-235 average enrichment, sintered UO_2 pellets, and 2) alumina wafers containing B_4C burnable poison for power flattening and longer core life. The additional 20 array locations are taken up by thimbles for guiding the stainless steel clad control rods. The 20 control rods associated with each fuel assembly are fastened together at the top by a spider-like bracket and are driven by a common, magnetic jack type control rod drive mechanism. Core life is 32,000 effective full power hours, giving a fuel burnup of 20,000 MWD/t and an operating period at 85% load factor between refuelings of over 4 years.

The propulsion plant consists of 2 normally cross-connected systems furnishing power to 2 shafts. Each main

engine is a marine-type, cross-compound, geared steam turbine rated at 37,500 SHP at rated steam conditions and exhausting to condensers at 27.5" Hg vacuum. Three stages of feedwater heating and conventional deaeration are provided. Two, 2,200 kw turbogenerator sets provide required propulsion plant and ship's service electrical power. A 2200 kw diesel generator and 2, 1,250 SHP electrical propulsion motors driving their associated shafts through their reduction gears are provided for emergency, take -home power.

6. The RCN and Rotterdam Dockyard 120,000 Nuclear Propulsion Plant -- (ref 85)

Following the August, 1969 ordering in the U.S. of 8, 30 knot, conventionally powered containerships of 120,000 SHP each, Reactor Centrum Nederland and the Rotterdam Dockyard Company undertook the design of a PWR nuclear propulsion plant for ships of this type. The design consisted of detailed extrapolation of Westinghouse (loop type) and earlier B&W CNSG (integral type) designs to 120,000 SHP and marrying these extrapolated designs to the ship with a minimum of ship redesign. The principal conclusions reached during this design were:

1) Both nuclear plants increased light ship weight 2400 tons due to heavier reactor and propulsion plant components and associated ship structural stiffening plus collision protection provisions, and 3000 more tons due to additional permanent ballast required for damage stability.

This total of 5400 tons extra weight is of the same order as the fuel supply weight for the conventional ship.

2) The larger-than-conventional nuclear propulsion plant requires more space, decreasing by 4 the total number of containers that can be carried by this ship. Minor rearrangement of spaces in the vicinity of the propulsion plant was necessary to realize their reduction of only 4 in container carrying capacity.

3) There seems to be no clearly decisive technical advantage for choosing one of these reactor types over the other. The choice of loop type reactor or integral type reactor would have to be made based mainly on differences in capital costs and operating and maintenance costs between the 2 (cost analyses of the 2 plants were to be done in later stages of the design and were not available to the author at this writing.). Generally speaking, however, the following comparisons of these 2 reactor types can be made:

a) Although on first appraisal the loop type may seem more complicated, its reactor vessel is smaller and lighter (107 tons weight vs. 275 tons) and generally easier to design and build. The steam generator tube bundle for the loop type is also smaller and easier to fabricate; heat transfer surface area is less than half that required for the integral type (16,800 ft^2 vs. 36,300 ft^2).

b) The integral type plant contains more primary coolant (3,530 ft^3 volume vs. 2,120 ft^3) at the same

average temperature (about 577F). Because of this volume difference, the design of the containment vessel to withstand complete expansion of the primary coolant following boundary rupture of the integral type plant results in either a much larger requirement for containment volume or in a higher end pressure after expansion in the smaller volume. On the other hand, the loop type requires considerably more containment space to provide room for thermal expansion of primary piping, tending to fix minimum containment volume independent of coolant boundary rupture considerations. If the integral type containment is to take full advantage of its inherent primary system consolidation, a vapor pressure-suppression containment vessel must be used. In compact form, the pressure-suppression containment consists of a dry-well connected with pipes to a wet-well. The wet-well, which condenses steam from the dry-well, also has the possibility of being designed to function as part of the radiation shield, further enhancing plant compactness. The connecting pipes in such an arrangement must be designed to direct steam underwater for any ship orientation (in case of ship sinking) and to prevent transfer of water from the wet-well into the dry-well during heavy weather.

c) For the 120,000 SHP plant the integral type pumps must be located on the top cover of the reactor vessel or inserted into the upper part of the vessel (location below the core tends to raise the center of gravity an undesirable amount). For the high flow rates involved, the

required net positive suction head of the pumps precludes self-pressurization (steam bubble in top of vessel), so that a separate pressurizer must be used.

d) Propulsion plant operating performance will be somewhat better for the integral type, since the somewhat superheated steam has nearly constant pressure at all power levels. By comparison, the steam pressure in the loop type tends to drop considerably (as much as several hundred psi) as plant power level is increased. Separate superheaters must be added to the loop type to obtain equivalent performance.

e) The integral type tends to maintain lower core temperatures during loss of coolant flow accidents, due to inherent natural circulation within the reactor vessel. Such natural circulation is more difficult to obtain in the loop type reactor design, although (as in the NERO design) it can be done.

7. Other Designs --

Many other nuclear marine propulsion plants have been designed, most of which appear somewhat less promising than those discussed above, at least for the near future. An exhaustive compilation of the details of all of these other plant designs would be much too lengthy to even consider for a work such as this. Accordingly, only a brief reference, by reactor type, to a few of these other plant designs will be included here.

a. PWR --

PWR ship propulsion plants have been designed by, among others, Mitsubishi Heavy Industries (Japan), Hitasi Shipbuilding and Engineering (Japan), Chanté de Atlantique (France), Ansaldo-Fiat (Italy), United Nuclear Corp. (U.S.), and the Dutch Atom Organization.

b. BWR --

BWR ship propulsion plants have been designed by, among others, General Electric (U.S.), American Machine Foundary (U.S.), Westinghouse Electric (U.S.), Société des Forges et ateliers du Creusot (France), Hetawerken (Sweden), Mitchell Engineering, Fairfield Shipbuilding and Engineering, and Combustion Engineering (jointly; all U.K.), and the United Norwegian-Dutch Atomic Center.

c. Organic Cooled Reactor --

Organic cooled nuclear ship propulsion plants have been designed by, among others, North American Aviation (U.S.), Atomics International (U.S.), the Polish Atomic Scientific Research Group (Poland), Hanover Technical University and Hamburg Atomic Energy Association (jointly; both W. Germany), and Hawker Sidderley Nuclear Power Co. (U.K.).

d. Gas Cooled Reactor --

Gas cooled nuclear ship propulsion plants have been designed by, among others, General Dynamics (U.S.), General Motors (U.S.), Ford Instrument Co. (U.S.), De Havilland Engine Co. (U.K.), Blom und Foss and Babcock & Wilcox (jointly;

both W. Germany), Simon-Carves Co. and General Electric (jointly; both U.K.), and Indatom Group (France).

IV. TOPICS OF SPECIAL INTEREST IN NUCLEAR MARINE PROPULSION--

Successful utilization of a reactor plant for merchant ship propulsion depends on the satisfactory solution of many engineering problems. Some of these problems derive from the basic necessity in the maritime industry to be economically competitive or go out of business. Others of these problems are more fundamental in nature and must be solved in order to have a technically acceptable, safe ship, regardless of whether the ship is economically viable or not. The basic difference in the safety problem between a nuclear ship and a conventional ship is that, for a nuclear ship, provision must be made to control, under all foreseeable circumstances, the radioactivity resulting from the fission process.

This section first discusses some of the economics involved in nuclear ship propulsion, including the more important problems arising from economic considerations. The section then discusses some of the more important fundamental problems; these latter problems tend to center around ensuring: 1) the continued health and well-being of those who operate the ship, and 2) continued reactor plant integrity and positive, long term containment of the highly radioactive fission products for the protection of the general public in the event of an accident. Finally, problems relating to plant maintenance and reactor refuelling are discussed. The pervading role of economic considerations in steering the solutions of many of these problems will be obvious.

A. ECONOMICS OF NUCLEAR VS. CONVENTIONALSHIP PROPULSION-- (ref's 86,87,90,91,95,96,97,98,103,106)

Existing literature regarding the economic competitiveness of nuclear vs. conventional propulsion for commercial ships abounds with confusing statements, unstated assumptions, and contradictory estimates. This, among many other factors, has resulted in the unfortunate state of merchant ship nuclear power in the U.S. today, a state that is characterized by a complex muddle of conflicting positions and near stagnation. If economic viability were the only factor involved, the way clear of this dilemma might be considerably less obscure. Complicating the economic decisions a potential nuclear powered shipowner/operator must make, however, are weighty Government administrative matters such as nuclear liability insurance, domestic and foreign licensing and ship construction and operating subsidies. These will be further discussed below.

1. Categories of Marine Shipping and Government Subsidies to the Shipping Industry --

Practically all of the commercial marine shipping done today can be divided into 3 main categories, each with its own distinct economic considerations. In addition, each trade route has associated with it conditions which can strongly influence economic viability within each of these categories. Failure to recognize and properly

account for these distinctions has contributed to the confusion generated by the many conflicting economic generalizations in the literature today. These 3 categories of marine shipping are:

1) General Cargo Ships -- Because the density of general cargo is low, hold volume is more important than deadweight capacity. However, the limited availability of general cargo normally makes full holds rare, accounting also for the generally small size of these ships. Since most general cargo vessels are in the liner trade, making scheduled pickups and deliveries, speed cannot normally be optimized, but must be adjusted to suit a logical schedule. Ship utilization factors (time in transit, as a percentage of a year) tend to be low. Machinery volume is generally more important than machinery weights. The above discussion must be modified for the containerships currently being built and operated. These vessels are generally large, high speed, fully loaded (because of dependable, faster delivery of cargo) ships with high utilization factors. It should be noted here that at least in this category it is an established fact that the fastest ship carries the cargo; this fact is the major reason for the recent, dramatic increases in containership speed capability.

2) Bulk Cargo Ships -- The practically unlimited availability of bulk cargo (oil,chemicals, ores, etc.) makes these ships most competitive when they have as large a dead-

weight capacity as possible. Since schedules are generally flexible, speed can be optimized. High ship utilization factors are common. The amount of added payload capacity to be gained from eliminating the need to carry propulsion oil on long routes depends on the maximum permissible vessel draft at the route's shallowest points (such as in canals or in port loading/discharging facilities). The productivity (tons of cargo delivered per year) of a given displacement, oil-fired, bulk cargo ship on a given trade route decreases above an optimum speed, as shown in Figure IV-1 below. The elimination of the need to carry large quantities of propulsion fuel oil in nuclear bulk cargo ships tends to result in productivity roughly proportional to ship speed.

3) Passenger Ships -- Like general cargo ships, these generally operate on a logical schedule such that speed cannot be optimized. Ship utilization factors are generally moderate, around 70-80%, but can be lower. Elimination of the need for propulsion oil can be most realistically exploited by reducing displacement while holding passenger and cargo capacity constant. Machinery weights are generally more important than machinery volume.

Another important economic consideration in deciding nuclear vs. conventional propulsion is the available amounts of Government subsidies and other financial aids. These were originally defined in the constitution of U.S. maritime policy, the Merchant Marine Act of 1936, and have recently

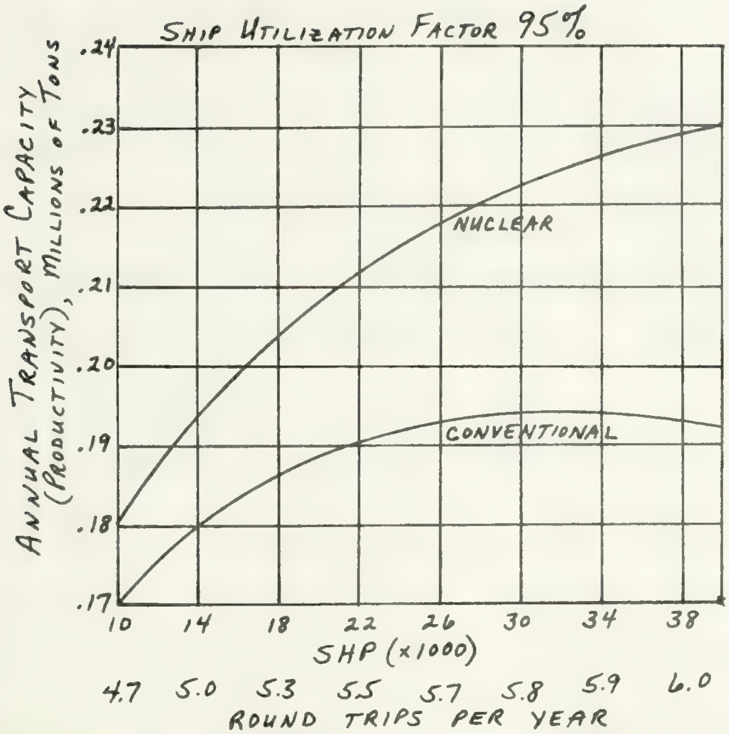

Figure IV-1 Comparative Annual Ship Productivity

been revised in H.R. 15424, The Merchant Marine Act of 1970. The 1936 law was passed because construction and operating costs had priced U.S. privately owned merchantmen out of competition with foreign flags. Recognizing that a national merchant fleet and shipbuilding industry are essential to U.S. defense posture, the law, as subsequently amended and as revised by H.R. 15424, basically provides a system of differential subsidies for construction and operation of cargo liners on certain essential trade routes, as follows:

1) Construction subsidy -- Originally limited to 55% of the construction cost for general cargo liners

and 60% for passenger ships, this subsidy is uniformly limited to 50% after 30 June 1970, 45% after 30 June 1971, and 2% less per annum until 1976 after which it is limited to 35%. Now also applicable to tankers and dry bulk cargo carriers, this subsidy is intended to enable U.S. shipowners engaged in foreign commerce to build ships in the United States on a parity with their foreign competitors. $143M was spent for this purpose in 1968.

2) Operating subsidy -- This subsidy applies only to ships operating on U.S.-foreign trade routes and not to those operating on domestic trade routes. The subsidy is based on the differences between the following costs comparing operation of the ship under U.S. vs. foreign registry: insurance, maintenance, repairs, wages and subsistence. Half of any profits in excess of 10% must be paid to the Government. $200M was spent for this purpose in 1968.

3) Tax benefits -- This provision allows shipowners/operators to deposit into a tax-free reserve fund both annual operational earnings and proceeds from the sale or indemnities from the loss of ships. The only requirement is that the deposits be used within a specified time for the construction, reconstruction or acquisition of other ships; accumulation of funds to gain interest is discouraged by the imposition of an interest charge (to be paid to the Government) on the reserve fund and by taxation of any earnings the fund might gain. As an example of the effect of this benefit, a company

in the 50% tax bracket can virtually double the vessels which it would otherwise be able to build.

4) Other supports -- Additional aid is provided U.S. shipowners/operators in the form of government insurance of commercial loans and mortgages for the construction or reconstruction of ships, direct loans at low interest rates for subsidized ship construction, and government acquisition of privately owned obsolete vessels if the purchase price is applied to construction or rebuilding of ships. Ships receiving these supports must be used on U.S. -- foreign trade routes.

2. Criteria for Economic Comparisons --

Complete validity of economic comparisons of nuclear vs. conventional propulsion plants requires that such comparisons be made within the frameworks of each of the 3 basic shipping categories, for specific trade route(s), and using consistent, meaningful criteria for comparison. In the marine shipping industry, such generalizations as, "Nuclear propulsion is (or is not) economically competitive with conventional propulsion," are meaningless. A much more meaningful generalization might be, "Any economic comparison of nuclear vs. conventional propulsion for one type ship on one trade route is generally not applicable to another type ship and/or another trade route." Given this, what criteria should be applied in such comparisons?

The basic criterion boils down to this: nuclear fuel savings must be capable of amortizing the higher nuclear capital and operating (other than fuel) costs in order for the nuclear ship to be economically competitive. That sounds simple enough. Application of such a criterion, however, reveals several, not easily quantifiable financial factors that could easily sway a shipping concern's decision to build a nuclear or conventional ship. Some of these factors are:

1) the presently lesser degree of predictability of the risk/return relationships for nuclear ships,

2) lack of an accepted rationale for assigning first-of-a-kind development costs to follow ships in a fleet of nuclear merchantmen,

3) possibilities of major cost-reducing break-throughs in nuclear plant technology in the near future,

4) lack of experience needed to accurately predict learning curve savings for multiple procurement of several nuclear ships,

5) uncertainty over opportunity cost of not building nuclear-propelled now, vs. catch-up costs later if the economic superiority of nuclear propulsion is proven by a competitor, and

6) uncertainty over the appropriate scrap value to assign the nuclear ship; for example, considerable expense at the end of the ship's useful life might be incurred in disposing of the reactor plant.

Furthermore, if comparisons of nuclear vs. conventional propulsion are to be realistic and accurate, they must take into account the many inherent differences in the economics of constructing and operating ships with these 2 types of propulsion plants. These differences include:

1) Capital cost -- nuclear tends to be higher due to: required framing and shell plating reinforcement to support and protect the reactor; more complex and exacting machinery layouts; use and effective segregation and control of more expensive materials, for example in the primary system and the shield (stainless steels, zircaloy, lead, etc.); the necessity to use more highly qualified (and paid) workers; and higher standards of cleanliness and quality control.

2) Operating cost -- nuclear tends to be higher in the areas of: higher amortization costs; nuclear liability insurance; radiological controls, chemistry and other worker training and qualification required for operation and maintenance of radioactive systems and components; a higher proportion of licensed personnel which may increase crew costs, including stores and supplies; and larger shore staff due to special safety and engineering requirements.

Even a superficial search of available literature on shipping economics will reveal a multitude of different criteria, each with its proponents and opponents. Included are such criteria as minimum shipping cost per cargo-ton mile, maximum annual profit, required freight rate, maximum

deadweight tonnage times speed divided by shaft horsepower, minimum average annual cost, discounted cash flow, maximum net present value, and capital recovery factor. Some of these criteria are valid and sound, while others tend to be invalid and tenuous. The most common failing of such criteria is failure to give proper weight to the time-value of money.

It should be noted here that use of 2 or more different criteria to compare the same alternate opportunities may likely lead to different conclusions. It should also be noted that more than purely economic analysis must be included in such comparisons. For example, on the basis of pure economics alone, sailing ships are even today the most "economic" form of ocean transport ever devised. As another example, ignoring factors of speed, range and deadweight capacity, the criterion of minimum shipping cost per cargo-ton mile indicates that oil-fired ships are for all cases more economical than nuclear, while inclusion of these factors can result in the opposite conclusion.

The traditional and most frequently used economic criterion in the marine shipping industry is the capital recovery factor (CRF), defined as annual gross profit divided by capital investment. Since this criterion does not account for the time-value of money, however, it relies on the alternatives having equal lives and on relatively constant profits over the life of the ships. CRF can be a valid criterion for deciding how to invest corporate funds (almost always the

situation in shipbuilding) only if certain conditions are met:

1) Excess funds not employed in the alternatives being considered may be invested at an equivalent rate of return; for the subsidized U.S. ship operator, this is not usually the case.

2) There are frequent, numerous opportunities to invest corporate funds; this is also not normally the case for U.S. ship operators, since several years may typically elapse between major investments adding to a ship fleet.

Such considerations have more recently led to selection of another criterion as more reliable: discounted cash flow (also called internal rate of return), defined as the present worth of cumulative revenues less cash outflows. From this analysis a rate of return is then calculated and used as the basis of comparison. The basic concept of such a comparison is that the best index of engineering success is profitability and the most meaningful measure of profit is the after-tax, net return on investment. Other considerations discussed above will, of course, still have an impact on the final decision.

Rather than include an exhaustive listing of past predictions of nuclear economic competitiveness or the lack of it, since these predictions tend to be confusing and contradictory anyway, this section will present a summary of certain conclusions of such predictions. The author believes these conclusions to be generally valid, unless

otherwise stated, for most nuclear marine propulsion application:

1) Because of the necessity that operational fuel savings and greater revenue earning capacity balance the incrementally higher capital cost, a nuclear propulsion plant must have a large power output to be economically competitive. This conclusion is borne out by typical curves of various criteria vs. power output, as shown in Figure IV-2 below. These curves are generalized to show behavior trends rather than absolute magnitudes. The high power requirement generally results in nuclear competitiveness only for such ships as high speed, moderate size, rapid delivery cargo ships carrying high-cost commodities on long routes, and moderate speed jumbo cargo ships carrying low-cost commodities but requiring high power because of their size.

2) The largest single factor tending to make the nuclear ship less competitive than the conventional ship is its significantly higher capital cost. Anything that can be done to decrease capital cost would enhance economic competitiveness; some possibilities include:

a) use of proven nuclear components and technology, thereby taking advantage of the development effort expended for other purposes, e.g. land-based nuclear central station electrical generating plants,

b) use of simplified fluid, mechanical, electrical, and control systems, and

c) use of ship structure to minimize foundation, containment and shield costs.

3) Realization of nuclear fuel savings requires a high ship utilization factor. This factor can be increased by decreasing port turnaround (loading and unloading) time; this in turn increases revenue by increasing the productivity (also called the annual transport capacity and defined as the tons of cargo carried per year) of the ship. Ship productivity is also increased for bulk cargo ships by elimination of the need to carry propulsion oil (15-20% of deadweight capacity on long routes).

4) Any reduction of nuclear fuel cost will enhance nuclear ship competitiveness. Some ways to reduce cost include:

a) increase fuel burnup,

b) simplify core design to reduce manufacturing and refueling time and cost,

c) minimize core size and equalize fuel burnup by increasing core power density and flattening the distribution of power generation in the core, and

d) increase ship utilization factor; interest charges (the cost of working capital) are proportional to the length of time the fuel is in the core. The effect of interest rate is approximately 0.1 mill/SHP hr for each 1% change in interest rate.

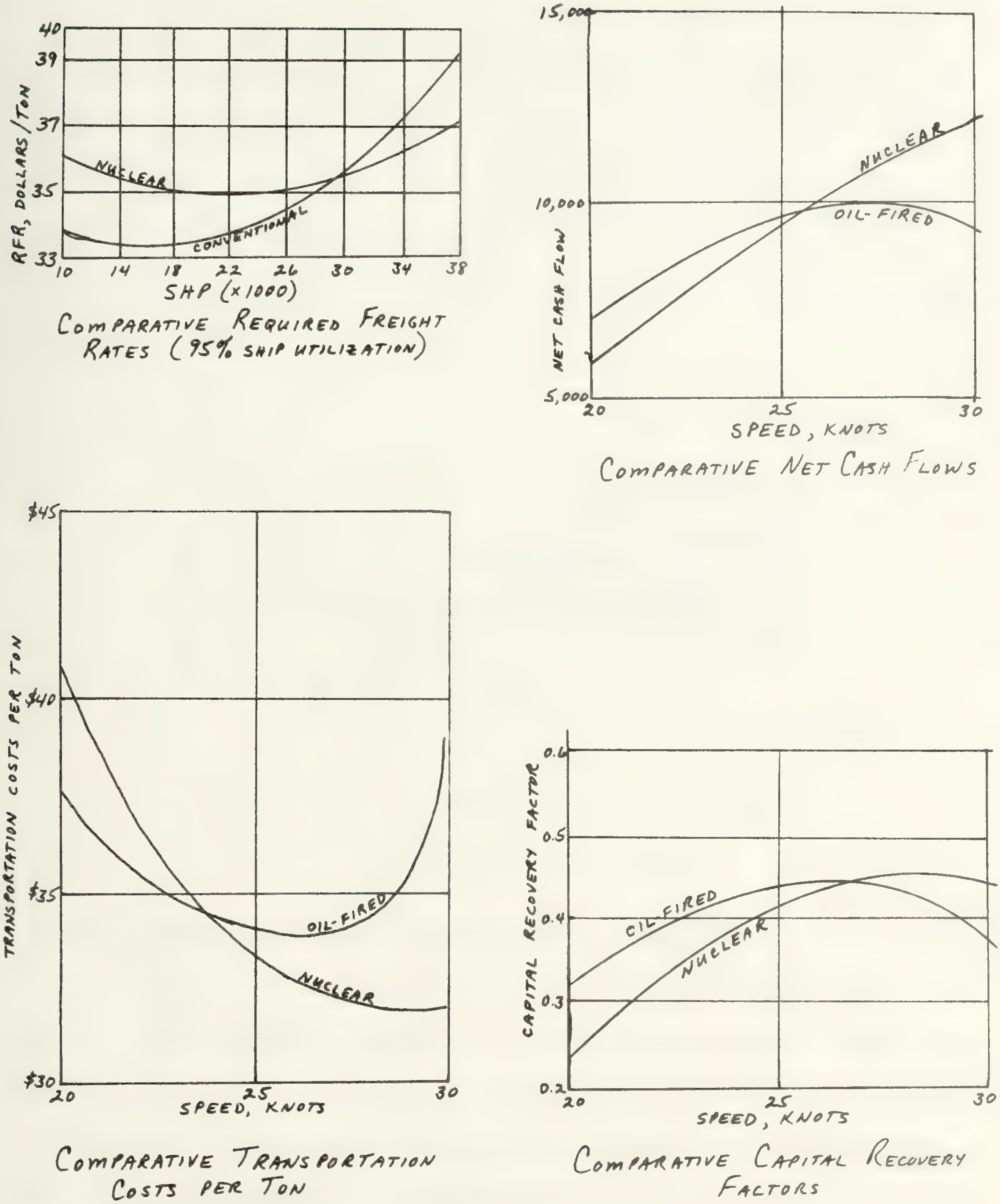

Figure IV-2 General Behavior of Selected Economic Criteria

5) Economic competitiveness of nuclear propulsion is strongly dependent on the fuel oil price assumed, as shown in Figure IV-3. "RFR" in this figure represents "Required Freight Rate"; the lower this value, the more economically competitive the ship tends to be. Fuel oil price has recently been steadily increasing worldwide, but varies widely from port to port.

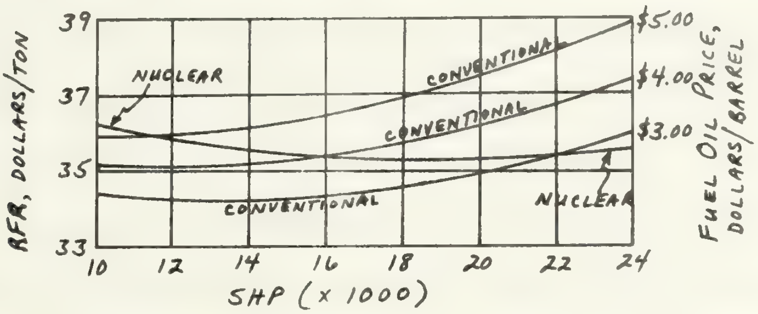

Figure IV-3 Effect of Fuel Oil Price on Nuclear Competitiveness

B. <u>RADIATION SHIELDING AND REACTOR SAFETY</u> -- (ref's 95, 96,99,16,100,101,102,86,81,24,103,107,108)

The primary, unique features of a reactor plant which make attainment of low weight, small size, and low cost far more difficult than for a conventional plant are the necessities to shield personnel from its potentially lethal radiation and to positively contain its prodigiously radioactive fission products under all conditions of operation and foreseeable credible accidents. This section discusses first some

shielding aspects of a reactor plant, then some containment considerations. Finally, some other aspects of reactor safety are discussed.

1. Radiation Shielding --

The basic problem in designing a reactor shield installation is to provide sufficient amounts of the proper materials to attenuate the various energy neutron and gamma radiations from core and coolant, while at the same time adding the minimum amount of weight and/or volume to the plant. Complicating the problem is the fact that the efficiency of various shielding materials for attenuating radiation is a function of both the type (neutrons or gammas) and the energy of the radiation. Suitable materials also vary widely in cost and physical properties.

To shield against fast neutrons, dense (high Z) material is needed. To efficiently thermalize intermediate energy neutrons, light (low Z) material is needed. To finally capture the thermalized neutrons, a material with large capture cross section, preferably one which releases a minimum of capture gammas, is needed. Shielding against gammas also requires a high Z material.

Obviously, no one material can satisfy all these requirements. A combination of appropriate materials must be assembled in a way which results both in satisfaction of

criteria regarding maximum allowable radiation levels (discussed in Section II above) and in a minimum weight/minimum volume/minimum cost shield.

Adding further complications to the shield design problem,these materials must be put together 1) to withstand thermal stresses due to radiation-deposited energy and mechanical stresses due to ship motions in a turgid seaway, 2) so that locally intense radiation "beams" do not occur where piping, electrical cabling and other penetrations must necessarily interrupt the shield, 3) to permit ready access to reactor components for maintenance and refueling and to containment vessel and critical support structures for routine inspection, and 4) to prevent unacceptable stress concentrations in the ship structure due to a too-rigid shield complex. Because reactor plant components themselves act as shielding and the coolant may be a source of radiation, coolant piping and component arrangement may have a significant effect on the shield design.

The most commonly used shielding materials are water, lead, iron and concrete. Each has its weaknesses, however: water needs a tank, which can corrode and leak; lead is structurally weak above 150F, requiring special (costly) installation methods, and is a source of capture gammas (up to 7 Mev in a thermal neutron environment; iron rusts and is a source of capture gammas (up to 10 Mev) in a thermal neutron environment; and concrete dries with 15%

void space, is thermal-stress sensitive, and needs cooling to avoid hydrogen loss. Other materials are also used, especially for certain specialized applications, such as boron in the water to enhance thermal neutron absorption and reduce capture gamma energy from 2.2 Mev to 0.5 Mev; boral (a sandwich plate mixture of B_4C powder and aluminum) to stop thermal neutrons on the shield outer face; and hydrogen-containing plastics, such as polyethylene, where no water tank is desired.

Overall shielding thickness tends to be fixed by 3 factors: radiation levels at the edges of the core and coolant; maximum allowable radiation levels outside the shield; and the radiation attenuation properties of the shield materials. The gamma intensity at the surfaces of a high-powered core is approximately ten billion (10^{10}) times greater than the biologically permissible level; for neutrons the ratio is about a trillion (10^{12}). Shielding decreases radiation intensity exponentially, rather than linearly. For example, a 2 in. thickness of lead reduces certain energy level gammas by a factor of 10; a 4 in. thickness reduces the intensity by a factor of 100; 6 in. by a factor of 1000. No amount of shielding, however large, reduces radiation levels to zero.

Fortunately, shield weight and volume savings are effected in the same way: by placing the shield materials as close as possible to the radiation source. On a weight

basis alone, all materials used in a reactor plant to attenuate gammas are very nearly equivalent. Geometry, however, gives a weight advantage to higher density material since it can be packed closer around the sources than a lower density material.

Detracting from the anticipated weight savings of using higher density materials is the additional structural material needed to support the concentrated weight and to supply the structural strength which most such materials lack. In actual designs, the use of lead reduces the weight of thick shield sections up to 20% as compared with steel; the designer must decide whether the additional expense of lead and its fabrication methods is warranted by such a weight reduction. Also, the degree to which this geometric effect can be taken advantage of is limited by the requirement that enough (secondary) shielding be placed on or around the containment vessel to limit personnel radiation exposure in the event fission products are released from the core into the containment vessel as a result of a reactor accident.

The effect of component arrangements, both inside and outside the shield, can be significant. Certain basic principles apply:

1) The most radioactive sources should be grouped close to the reactor, with the less radioactive components outside them affording some shielding.

2) All radioactive sources should be kept as

small and as low in the ship as possible.

3) Penetrations through the shield should be located and/or designed so as to require the smallest amount of additional, compensating shielding.

4) Maximum permissible radiation levels should be increased outside the shield, where possible, by making spaces immediately outside the shield consist of tankage, store-rooms, or other seldom-occupied spaces.

2. Reactor Safety and Radioactive Material Containment --

The subject of reactor safety is a very broad one, encompassing a wide range of topics from location of the reactor in the ship to core physics design for maximum inherent reactor self-control, to selection of shore facilities and training of plant operators. Experience to date with both stationary and mobile reactor plants has demonstrated that the probability and the consequences of a serious nuclear accident are extremely small providing the design is adequate, the equipment is sound and properly installed and maintained, the personnel are properly trained, and proper operating procedures are followed.

In any field of human endeavor, however, it is inevitable that mistakes will be made; accidents will occur. For example, in spite of profuse safety precautions in effect, shipboard fires and explosions still happen rather frequently;

propulsion is lost and ships founder; and ships collide with each other and with fixed installations. Expensive equipment and human life are lost. The risk involved in ship operation are generally known and reluctantly accepted based on a large background of experience.

Reactor accidents, however, tend not to be viewed with such consistent degrees of objectivity and acceptance; the reasons for this are many. In spite of the variety of reactor accidents that have happened to date, the consequences of any given reactor accident are not generally predictable with a great amount of certainty. The path is long and complex between adequate, positive containment of hazardous fission products and their potential release and the disastrous effects such release could have on the ship operators and the general public. There are many important variables which might act separately or together in a favorable or unfavorable manner. The resulting outcome of a reactor accident could range from negligible damage to widespread release of radioactivity costing hundreds of millions of dollars to clean up and resulting in accurately unassessable damage to affected people and their offspring. The extremely complex, emotional and costly court cases that would inevitably result from the latter, along with the potentially serious setback such an incident could give other practical applications of nuclear power, can be imagined.

Small wonder, then, that reactor safety standards evolving from sincerely motivated engineering analyses of potential reactor accidents tend to be very high; since such analyses inevitably demand more information than is usually available, the assessor would be less than human if he did not err on the side of pessimism and conservatism. The safety of a nuclear propulsion plant is affected by many variables, from the basic selection of reactor type (PWR, Gas cooled, etc.) to the details of material control and fabrication techniques used in plant construction. Engineered safeguards (including those built into the ship as well as the reactor) play an important role; one of the more important of these safeguards is containment of radioactive material resulting from reactor operation.

Reactor plants for power production generally include 3 barriers to the release of radioactive fission fragments; maintenance of the integrity of each of these will be discussed below:

1) the fuel cladding, designed to prevent release of fission products to the coolant; note that another barrier, the fuel material itself, is sometimes included in such a listing if the material is relatively non-corrosive when exposed to the coolant,

2) the primary coolant external boundary, such as reactor pressure vessel, coolant piping, and the primary pressure restraining walls of primary components such as

coolant pumps, steam generators, control rod drive mechanisms, valves, and the pressurizer, and

3) the reactor containment vessel, designed to contain all primary coolant, and any radioactive material entrained in it, in the event of a major rupture of the primary coolant boundary. Although this barrier can utilize integral portions of the ship's structure, a separate structure is usually cheaper and lighter due to the inordinately large scantlings required to adequately stiffen normal ship structure to withstand pressures associated with certain reactor accidents. Note that another barrier, the ship's hull, is sometimes included in such a listing, since it could effectively delay release to the environment of any fission products which might leak out of the containment vessel.

The integrity of the cladding is a function of the selection of fuel and cladding materials, cladding and coolant materials, and the chemical and metallurgical compatibility of these 2 sets of material systems. Cladding integrity also depends on proper selection and maintenance of coolant chemistry conditions (pH, ion content, etc.) and on the ability of the plant to maintain cladding temperature below certain critical values by controlling fuel temperature to remain below certain upper limits and maintaining adequate heat removal capability. The latter (including decay heat removal) will be discussed below. One additional important variable is the compatibility of the cladding material and sea water;

this could be important if a ship sinks in deep water with its primary coolant boundary breached.

Once the plant is designed and built, it is up to the trained operators to periodically check and maintain the prescribed coolant chemistry, to operate the plant within the specified parameters of temperatures, pressures, flow rates, etc., and to perform periodic checks and preventive maintenance on plant components, including indication, control, and safety systems, as required to ensure proper operation.

Effective control of fuel and cladding temperature requires maintaining a balance between heat generation rate in the fuel and heat removal rate in the coolant. Heat production is increased by inserting reactivity into the core (such as by control rod withdrawal, moderator temperature change, or void fraction change). Both design and operational attention is required to prevent accidental increases in heat production. Control rod drive mechanisms must operate normally at all feasible ship inclinations, be unaffected by ship motion, and remain fully inserted with the ship upside down; they must also be capable of reliable, rapid rod insertion at all angles of ship inclination. Reactor control and safety systems must be adequately designed and conscientiously maintained by periodic checks and alignments in such a condition as to reliably actuate the control rods to insert at normal, faster than normal, or scram speeds when necessary. Approved plant operating procedures must be rigidly adhered

to in order to prevent accidental reactivity insertions such as due to inadvertent introduction of rapid coolant temperature and void changes due to improper equipment operation.

The integrity of the primary coolant boundary is a function of the selection and compatibility of coolant and boundary materials, the adequacy of boundary materials and fabrication processes to withstand steady and varying stresses, such as those due to pressure, to temperature changes, and to inertial forces in a seaway, throughout plant life. Adequate design allowance for such processes as corrosion and radiation embrittlement must be included and -- just as for cladding -- the prescribed coolant chemistry must be maintained. In addition, the secondary (working) fluid chemistry must be maintained within prescribed limits in order to preclude steam generator tube failure.

Since the reactor containment vessel is an "if all else fails" type of protection, its integrity must be maintained even if in the most severe accidents (collision, grounding, breaking up, fire, etc.) the reactor and possibly the entire ship is a total loss. Intended to protect the general public, the presence of the containment vessel must not be allowed to give rise to an exaggerated sense of security or a reduced effort in developing safer reactors. The integrity of this vessel is maintained by periodic nondestructive testing, including pressure drop testing, and by providing protection against impact such as that due to

collision and grounding. Typical protection might consist of reinforced ship bottom and side structure in the vicinity of the reactor plant; current rules, based on scale model tests done in U.K., U.S., W. Germany, Japan, and Italy, require the minimum thickness of such structure to be Beam/5, or 10 ft, whichever is larger.

Penetrations through the containment vessel deserve special attention in the design since these normally constitute the main source of leakage. In general, ventilation penetrations must either be avoided or kept small and specially designed for leak-tightness; personnel access hatches must be arranged and designed so as to maintain containment vessel integrity and/or be capable of immediate, automatic closure to regain integrity in the event of an accident during periods of personnel access to the vessel with the hatch open. Piping penetrations must be individually examined to determine whether or not there is a need for additional valves or other system provisions to ensure leak tightness in the event the piping ruptures external to the containment vessel. Electrical penetrations must be specially designed in order to be able to withstand higher than normal pressures and temperatures in a reactor accident without leaking.

The design pressure of the containment vessel depends on the type and detailed design of the reactor plant. The tacit assumption is generally made, because of the very high degrees of conservatism in its design and care in its

construction, that the reactor pressure vessel will not rupture. Instead, it is assumed that the maximum credible boundary rupture accident would consist of the largest penetration(s) through the reactor pressure vessel instantaneously being opened up, such as by a double-ended rupture of the attached piping. Flashing and/or expansion of the coolant as it leaves the opening and/or consequential exothermic chemical reactions result in a pressure buildup to a peak, followed by a pressure decay as the coolant condenses and/or cools. The containment vessel must be capable of holding this peak pressure with an extremely small leakage rate. Various types of pressure suppression systems, including wet-well/dry-well vapor condensers, spray headers, cooling fins, and cooling coils, can be installed to reduce the magnitude of the pressure peak and more quickly return containment vessel internal pressure to atmospheric.

The design of collision and grounding barriers for protection of the containment vessel and its contents is usually based on a principle widely used in the armaments industry: kinetic energy of impact is absorbed to the maximum extent when as large a volume of the barrier as possible is yielded, i.e. deformed through and beyond its elastic limits.

The likelihood of collision and grounding accidents can be significantly reduced, moreover, by installation and effective use of available, good quality navigation equipment,

radar/computer anti-collision devices, and signalling and communications equipment, by having adequate astern power, by provision of such safety features as automatic changeover to emergency steering gear upon failure of the normal gear, and by utilization of available weather track routing schemes. In addition, provision should be made to prevent collapse of the containment vessel due to sea pressure in the event of deep water sinkage of the ship; such a provision might typically consist of one or more pressure-equalizing check valves which would admit sea water to the containment vessel below a certain depth but prevent escape of vessel contents.

C. REACTOR CORE DECAY HEAT REMOVAL --

As discussed above, fuel cladding integrity depends on keeping its temperature below melting by maintaining a balance between the heat generation rate of the fuel and the heat removal rate of the coolant. This section discusses the problem--inherent in the design of all reactors--which generally gives rise to the most difficult heat transfer criteria the reactor must meet: removal of core decay heat in accident situations. Section II above indicates the amount of heat generated by radioactive decay of fission products in the core following reactor shutdown. A typical value of decay heat after shutdown following several hours of full power operation is about 3% of rated power. For SAVANNAH this corresponds to 2,000 KW; for the 120,000 SHP

CNSG plant, 9,000 KW. If this heat is not removed, burnout (melting of the cladding and possibly the fuel itself) can occur within seconds to minutes, depending on specific values of many parameters of the reactor plant.

Heat removal rate can be severely reduced in a number of ways, such as:

1) Loss of coolant -- typically could result from rupture of a part of the primary coolant boundary and release of up to 90-95% of the coolant into the containment vessel. With effectively no medium for the heat to be transferred to, its energy is applied to heating up the fuel elements; radiative heat transfer becomes effective at temperatures too high to prevent fuel element damage. As noted above, hazards analyses for loss of coolant accidents generally make the assumption that although any other portion of the primary boundary may rupture, the reactor pressure vessel remains intact.

Fuel cladding integrity following such an accident can generally be assured if the core can be kept sufficiently covered with coolant. To this end, many designs locate all reactor vessel penetrations above the core (e.g. MUTSU, CNSG, UNIMOD, and the Westinghouse designs described in Section III above). In addition, emergency injection systems are often provided; these systems direct coolant into the reactor vessel or directly onto the fuel elements following a loss of coolant accident. These schemes can be rendered ineffective,

however, if extreme ship angles result in uncovering of the core or if total loss of power prevents actuation of the injection system(s).

2) Loss of coolant flow -- typically could result from loss of power to the coolant pumps or circulators, resulting in rapid flow rate decrease; such an accident in a marine plant must be considered a likely event. Final value of flow rate through the core in this accident depends on amount of natural circulation capability designed into the plant. Coolant pump inertia may be very important in precluding core damage in the early stages of this accident. As the (nearly) stagnant coolant heats up, fuel element temperatures increase. Fuel element temperature rise in a water-cooled core can be aggravated by steam blanketing and subsequently larger coolant to fuel element temperature differences due to the lower heat transfer coefficient of steam.

Fuel cladding integrity following such an accident depends on maintaining or quickly reestablishing adequate coolant flow rate. The surest scheme, perhaps, is natural circulation flow capability, such as in the BWR plant or in the NERO design described above, since this does not rely on any source of power for maintaining coolant flow; extreme ship angles, however, could greatly inhibit or even prevent natural circulation of coolant. Backup power supplies independent of the reactor, such as diesel generators or

batteries and motor generators, can also be used to reestablish flow. In the event the plant must be cooled down with limited power available, secondary steam can be bled from the steam generators until coolant temperatures and pressure are low enough to actuate a separate, emergency coolant heat exchanger. Alternately, once the decay heat rate is sufficiently low, coolant purification system heat exchangers could be used.

3) Loss of coolant pressure -- typically could result from a small coolant leak or from pressurizer or pressure relief valve malfunction. As the effective density of coolant available to remove heat decreases due to expansion (e.g. for gas coolant) or steam void formation, thereby reducing the heat transfer coefficient, the temperature difference between fuel cladding and coolant rises. This, coupled with higher coolant temperatures, results in increasing fuel cladding temperatures. Coolant charging or injection systems and/or alternate means of pressurizing the coolant, such as use of compressed gas in the pressurizer, can be effective in maintaining fuel cladding integrity.

4) Loss of a heat sink -- All of the above accidents can be mitigated by rapid provision of a sink for the decay heat, whether the sink be latent heat of vaporization in the steam generators or sea water via another heat exchanger(s). The potentially disastrous consequences of wholesale core meltdown make it mandatory that the normal heat sink (secondary working fluid) be backed by at least one alternate sink.

The shipboard installations and other reactor plant designs described above give examples of some of the many possible alternate heat sinks that can be used. Dry docking periods for a nuclear ship are particularly susceptible to accidents resulting in loss of heat sink; in such periods, special decay heat removal provisions, such as sea water cooling hoses with backup hoses and/or backup power supplies for all pumps may be required to ensure adequate core cooling.

D. REACTOR REFUELING AND SERVICING FACILITIES --

Refueling of a nuclear propelled ship is likely to be the most hazardous operation of its entire service life. In the refueling operation the number of barriers between the fission products and the general public is reduced from 3 to only 1. Containment and reactor pressure vessels are opened for refueling access, and fission products are contained only by fuel cladding. Fission product inventory and decay heat generation rate tend to be large, so adequate shielding and cooling must be provided. Obviously, specialized facilities and equipment are required. This section describes the refueling operation and some of these facilities.

Exact details of the refueling operation depend, of course, on exact details of the reactor plant design, so a typical refueling sequence of a PWR plant will be described in rather general terms. Preparations prior to arrival of the ship involve writing and/or checking out step-by-step

refueling and emergency procedures, designing and manufacturing and/or checking out special tools and equipment, and training and checking out the personnel who will perform the refueling operations. It is difficult to overstate the importance of thoroughness in these procedures and of a strictly enforced requirement for non-deviation compliance with them by personnel performing the refueling operations. An ample supply of calibrated radiation monitoring instrumentation, protective clothing and other ancillary equipment must be on hand.

Ship arrival is followed by calm water docking or, preferably, dry docking so as to keep relative motions between reactor and crane to a minimum. All explosives and fire hazards should be removed from the vicinity prior to opening the containment boundary. Deck access hatches and the containment vessel refueling port are removed. The plant is cooled to near ambient, using decay heat removal or other systems, and depressurized. Reactor vessel coolant level is then lowered below the vessel head flange. The real radioactive work begins with reactor dismantling. Radiation control/cleanliness control areas are set up and full protective clothing donned. Control rod drive mechanisms are disconnected and removed. The reactor vessel head is unbolted and removed and a cylindrical neck is fitted to the vessel flange. This extension is filled with purified water and underwater lights and viewing equipment are

installed. Control rod shafts are removed into shielded containers and stored. Reactor internal fittings such as coolant flow baffles and core hold-down devices are removed into shielded containers and stored. Direct access to the fuel elements is now possible.

The spent fuel assemblies are withdrawn from the reactor vessel, precisely and with great care to prevent damage to the single remaining fission product barrier, into a lead-lined steel handling cask. The cask has its own decay heat removal system to keep the fuel from overheating, plus a lead-lined bottom plug to complete the shielded enclosure. The crane transfers the cask and its contents to a heavily shielded, cooled shipping cask mounted on a railroad car. Alternatively, the assemblies may be stored in a water-filled pit for further decay before shipment. Removed fuel assemblies are later shipped to a fuel reprocessing facility for recovery of the fissionable material in them and disposal of the remainder of the assembly. After remote inspection of the remaining vessel internals and replacement of any defective parts, the reactor is ready for insertion of new fuel.

New fuel insertion must be an extremely well-controlled process in order to ensure that each fuel element is placed in the correct location. Incorrect placement of fuel with varying enrichment or burnable poison concentration can result in irreparable, potentially disastrous fuel element

damage when the plant is operated at high power. Following fuel installation, the reactor is reassembled essentially in the inverse order of its disassembly. The entire refueling area is then carefully decontaminated and the containment vessel is resealed.

In addition to refueling, a nuclear ship may require periodic overhaul of certain components, such as reactor coolant pumps, and occasional repair or replacement of maloperating or defective plant components. Since these components may be highly radioactive, the reactor servicing facility must have the equipment and trained personnel needed to remove, work on, and replace such components.

Moreover, nuclear ship operation generates solid and liquid radioactive wastes that must be periodically disposed of. These wastes vary from low radioactivity level items such as rags, disposable gloves and periodic coolant samples drawn for chemistry analysis, to high radioactivity level items such as expended coolant purification system ion exchanger resins.

Minimum facilities, then that must be provided by a reactor servicing facility include:

1) Cranes and other necessary transporters heavy enough to handle large reactor plant components and shielded casks involved in refueling/reactor servicing operations; these should be adequately tested prior to use to preclude the potentially disastrous consequences of dropping one of

these units, possibly even onto the reactor itself,

2) Special tools and equipment needed for refueling /reactor servicing operations; included are such items as remote viewing and handling equipment, machine shop facilities for both radioactively contaminated and uncontaminated work, shielded handling and storage casks, reactor vessel extension pieces, and special lifting and rigging gear,

3) Mockup areas for checking out procedures, equipment and personnel prior to ship arrival,

4) Special storage and handling areas for unused fuel elements with adequate provisions for avoiding inadvertent assembly of the fuel into a critical mass,

5) Decontamination facilities to remove radioactive contaminants from reactor plant components requiring servicing, thereby reducing radiation levels and minimizing spread of the contaminants,

6) Decontamination facilities for cleaning and processing radioactively contaminated clothing, tools and other items, including personnel,

7) Special storage and handling areas for spent fuel storage and decay prior to shipment for reprocessing and disposal,

8) Radioactive waste transfer, storage and processing facilities, and radioactive solid waste transfer, processing, packaging and storage facilities; wastes handled would originate from both the ship and the facility,

9) Storage areas for spare reactor plant parts and equipment not carried aboard ship, and for the special tools and equipment needed for refueling/reactor servicing,

10) Health physics facilities with adequate radiation instrumentation and trained personnel for all refueling/reactor servicing operations, and

11) Office spaces for required engineering and administrative personnel.

V. FUTURE DEVELOPMENTS

(ref's 32,59,74,91, 103, 105, 109 through 126)

As the foregoing sections indicate, nuclear propulsion is today, and has been for many years, technically feasible for many different types of ships. The economic competitiveness of nuclear propulsion, however, has been for many years, and still is, a very complex issue with many variables, including ship type, SHP and trade route. In addition to the technical and economic considerations involved in discussions of commercial propulsion, a third factor must be included for completeness; that factor is political considerations.

Any attempted forecast of the future applications of nuclear propulsion to merchant ships runs a grave risk of landing wide of the mark if the political considerations involved are overlooked. Accordingly, this section first discusses these political considerations in an historical framework. Various types of ships are then examined regarding their suitability for nuclear propulsion. Finally, the current status of the economic viability of nuclear propulsion is discussed and a forecast of future applications is presented.

A. POLITICAL FACTORS IN COMMERCIAL NUCLEAR PROPULSION --

Neither a thorough explanation of why the U.S. today has no nuclear merchant ships nor a meaningful forecast of future nuclear propulsion marine applications can be made

in isolation from the history of past such applications and attempted applications. A cursory review of pertinent events from the early 1940's to the late 1960's is included to provide this history.

Following an historically recurrent pattern of postwar maritime decline, the United States entered World War II with a merchant marine totally inadequate to the logistic need of the war. A viable merchant fleet was constructed on a crash basis at a staggering cost. Following the war, the usual sacred oaths were taken that the U.S. would never again permit such a maritime situation to exist. Table V-1 indicates how little effect this experience and these oaths had toward maintaining a viable U.S. merchant marine. In the short period of time from 1950 to 1966, the U.S. active flag fleet, consisting of privately owned ships and government vessels not in the National Defense Reserve Fleet, dwindled from 14.1% of the world fleet to only 6.8%. While the total tonnage of U.S.-foreign trade more than doubled, in this period, the amount of total tonnage handled by U.S. flag ships dropped by more than half. Table V-2 shows the remarkable growth of other merchant fleets in this same time period.

Table V-1 Postwar Decline of the U.S. Merchant Marine; numbers in parentheses indicate percent of the world fleet.

	1950	1966
Tankers	457(21.5%)	277(7.7%)
Bulk Carriers	54(9.6%)	57(2.7%)

General Cargo Carriers	1049(13.6%)	804(7.5%)
Passenger/Cargo Ships	57(5.2%)	29(3.4%)
Totals	1617(14.1%)	1167(6.8%)

Table V-2 Postwar Growth of Certain Other Merchant Marines

	1950	1966
Japan	387(3.4%)	1405(8.1%)
U.S.S.R.	432(3.8%)	1343(7.8%)
Norway	959(8.3%)	1356(7.8%)
W. Germany	174(1.5%)	860(5.0%)
Greece	218(1.9%)	952(5.5%)

The U.S. merchant marine just 2 years ago consisted of 115 ships, over 90% of which were over 20 years old. These ships carried only 7% of the total U.S.-foreign waterborne import and export trade. Ship construction for this trade was only 12 ships per year. By contrast, the U.S.S.R. merchant marine just 2 years ago consisted of over 1400 ships, 58% of which were less than 12 years old. Russian ship construction for foreign trade was 100 ships per year. Just one of the regretable aspects of this decline is the large balance of payments deficit attributable to the insufficiency of the U.S. merchant marine (i.e., the difference between payments and receipts for ocean shipping of U.S.-foreign trade): over $500 million per year.

The U.S. merchant marine is unable to compete in the world market today for the same reasons it could not in 1936: basically, high costs. Very simply, it costs from 2

to 3 times as much to build and operate a ship under the U.S. flag as under foreign registry; most of this difference is attributable to the higher cost of U.S. labor. Other contributing factors include: inflexibility or unsuitability of equipment, obsolescent vessels, inefficient port operations, rigid and antiquated trade routes and regulations, and labor instability. Moreover, the subsidy system set up in 1963 has only served to prolong the decline of the U.S. merchant marine, while not contributing to the solution of its more basic problems.

More importantly, the several government agencies, shipowners, shipbuilders, maritime labor and all manner of merchant marine interests have failed to agree collectively or individually how to solve these maritime problems. Proposals and counter-proposals have been seemingly endless while the exigency of the situation demands that they not be endless. As each year of maritime indecision and uncertainty passes, the U.S. merchant marine declines and the U.S. becomes more and more dependent on the ships of other nations to carry its oceanborne trade.

The destiny of U.S. merchant ship nuclear propulsion is controlled principally by 3 groups: government, nuclear industry, and shipowners. The roles and interests of each of these 3 groups are instrumental in the success or failure of U.S. maritime nuclear propulsion and will be briefly reviewed.

1) The Government -- From the earliest days of the SAVANNAH, the government's attitude toward commercial nuclear propulsion has been divided -- as has been its attitude toward the problems of the U.S. merchant marine since the early 1950's. President Eisenhower's announcement of the new "Peace Ship" took the Democratic Congressional leadership by surprise and aroused a skeptical, if not hostile, response from those on the Hill who considered the ship to be a partisan political initiative by the Republican administration. Following SAVANNAH's launching and early operation, many realized that nuclear propulsion could play a most important role in upgrading the U.S. merchant marine; in fact, many leaders in both government and industry were firmly convinced that nuclear propulsion was the only remaining chance for the U.S. to regain a competitive position in oceanborne shipping.

Various ideas were aired, by various agencies, companies and other groups, for follow-on merchant ships. Some argued that construction and operation of a fleet of 4 or more commercial nuclear ships would be the only convincing way to determine the true cost of nuclear ships using current technology, and the only sure way to determine the most fruitful areas to apply development effort to in order to reduce costs in more advanced nuclear ships. This fleet was envisioned as fulfilling the same role in nuclear ship development that the Yankee and Dresden stations did in central station nuclear power; the civilian nuclear power

development program was not initially economical either, but is now considered an economic success. When counter-arguments were advanced based on lack of nuclear economic competitiveness and the desire to develop an optimum plant prior to further shipboard applications, attempts were made to justify construction of follow-on nuclear ships based on such arguments as ultimate long range economic benefits, national prestige, and the strategic value to the U.S. of having a fleet of high speed nuclear cargo ships capable of serving the needs of the Armed Forces overseas in any emergency.

In addition to these ideas, various forms of merchant marine reform legislation were proposed year after year. This proposed legislation included a variety of measures intended to alleviate the more basic merchant marine problems, while giving strong support for construction of additional nuclear merchant vessels. Speeches were given, testimony was presented, political pressures were applied and met with counter pressures, and so it went; but the "accountants", who insisted the next nuclear ships must be strictly economically competitive to be politically acceptable, and the "testers", who insisted the next nuclear marine PWR must be thoroughly researched in a land-based prototype, won out. Nothing substantial emerged.

The fate of follow-on nuclear ships remained tied to the larger problem: failure of the Executive Branch to

formulate a meaningful overall program to revive a declining U.S. merchant marine. Underlying this problem is the fact that practically any effective program for restoring the U.S. merchant marine to a position of world leadership, whether or not nuclear ships are involved, would involve expenditure of large amounts of money; this would infringe on other national priorities and could require cutbacks in other, more politically favored programs. Compounding the situation, jurisdictional frictions within the Executive and Legislative Branches prevented effective action to get follow-on nuclear ships built independent of the larger merchant marine problem. The AEC and the Joint Committee on Atomic Energy showed reluctance to relinquish control over a project which logically should have passed over into those agencies of the Legislative and Executive Branches which have primary responsibility for commercial merchant marine affairs (the House Merchant Marine Committee, the Senate Commerce Committee, and the Maritime Administration).

For example, following termination in 1964 of its Maritime Nuclear Program, the AEC set up in 1965 the requirement for extensive AEC research effort (including construction of a $35,000,000 prototype) as a prerequisite for approval of the current Maritime Administration proposal for follow-on PWR-powered ships (this requirement was set up in spite of extensive PWR experience gained in both the naval reactors and civilian nuclear power programs, including that with SAVANNAH herself, which had no prototype). By 1968

the AEC had virtually closed out its interest in merchant ship nuclear propulsion and in its 1965 requirement for a PWR ship propulsion prototype.

During this period no clearcut policy was developed regarding extension of subsidies and other aids, including nuclear liability insurance coverage, to follow-on nuclear ships; the Maritime Administration (MarAd) preferred to await AEC plant development before broaching these issues. Nor were any agreements reached for cooperation among the Maritime Administration, the AEC, shipowners and nuclear industry regarding the construction and operation of follow-on ships.

2) Nuclear Industry -- In the early 1960's, following significant success in the application of nuclear power to central station electric power plants, U.S. reactor manufacturers were considerably interested in the potential maritime nuclear market. Several detailed designs for follow-on reactor plants were developed, some of which are described in Section III above. By 1965/1966, however, the lack of a clear-cut, cohesive government policy regarding follow-on ships and the obvious AEC and MarAd feeling that an optimum, fully economical maritime nuclear plant should be developed before proceeding with practical commercial application had discouraged most of the industry into channeling its resources into the expanding, more lucrative nuclear central station market. Only Babcock and Wilcox

and Westinghouse continued to expend significant amounts of effort into the late 1960's toward developing a marine reactor plant.

3) Shipowners -- Ship operators, unlike ship buffs, make a sharp distinction between technical feasibility and commercial practicality. Confronted with today's shipping problems, shipowners have tended to wrestle with analyses of fossil steam vs. diesel and possibly even gas turbine, while dismissing nuclear propulsion as a luxury they can't afford without large amounts of government aid. In addition, the general experience of most shipowners has convinced them that changing from a known, satisfactory design to a new design of any item of machinery brings the risk of loss of availability time and unexpected costs for maintenance. Accordingly, shipowners have tended to change machinery types only if they are unsatisfied with the present type's reliability and/or relatively certain that significant economic advantages will outweigh the risks in the new machinery type.

In spite of this, at least 2 large shipowners had studied in depth and proposed to the Maritime Administration construction and operation of a follow-on fleet of 3 or 4, high speed, long route, high utilization containerized cargo ships. Not able as private operators to assume the total burden of a nuclear ship program, these companies offered in the mid-1960's to individually fund construction of the

necessary port facilities and containers and to contribute to the ship construction cost the same amount they would have paid for conventional ships of the same capabilities. The lack of a clear-cut government policy precluded action on any of these proposals and the ships were built in both cases with conventional power plants.

U.S. shipowners have in the past pioneered many maritime developments, including those listed below, but the large capital outlay required for a fleet of second generation nuclear merchantmen with only little government aid certain was considered by them to be beyond limitations dictated by common business prudence. Some of the developments pioneered by U.S. shipowners are:

a) the 16,700 dwt T-2 tankers (called, when built, too big for practical commercial use),

b) postwar tankers of 28,000 dwt (called, when built, supertankers with limited usefulness),

c) the 12,900 dwt, 20 knot Mariner class cargo ship (said, when built, to be too big and costly for commercial service, and

d) containerships.

B. SUITABILITY OF SPECIFIC SHIP TYPES FOR NUCLEAR PROPULSION --

As discussed in Section IV, economic viability of nuclear propulsion generally requires high power/high speed,

a long trade route, and a high ship utilization factor. In the 1930's American merchant ships steamed at 8-10 knots. World War II-built ships (which still make up over 75% of the American merchant fleet) steamed at 12-16 knots. Postwar-built ships steamed at around 18 knots, while the average ship built today steams at 20-24 knots.

This increase in speed has been made in response to the needs of shippers who want to minimize the length of time their high value goods are in transit in order to expedite payments from customers. Since required fossil fuel capacity increases, for a given trade route and ship size, as the cube of ship speed, the trend in ship size has also been upward in order to minimize the net reduction of cargo carrying capacity due to the larger fuel oil capacity required.

Keeping pace with the increase in ship size and speed has been the increases in oceanborne cargo. In dollar values, free world oceanborne trade is today approximately $370 billion annually; although 20% of this is American cargo, less than 3% of it is carried by the American merchant fleet because 1) the bulk of its ships are simply not competitive with the more modern, technologically advanced foreign fleets and 2) its shipping capacity has been steadily decreasing since the end of World War II. In tonnage value, oil tankers now transport about 1500 million long/tons annually, about 60% of the total oceanborne trade; this figure is growing

about 8% per year and will require 150 million tons of tanker capacity by 1980. World dry-cargo trade, growing at 4.5% per year, will require 175 million tons of cargo shipping capacity by 1980. Two major changes have recently taken place due to this growth in ocean borne trade:

1) Seaborne transportation is achieving dramatic economies of scale. For example, carrying bulk cargo from Los Angeles to Australia in a 130,000 ton ship costs about the same as carrying it 200 miles by rail in the U.S.; a 150,000 ton tanker moves crude oil 5,000 miles at less than 1/4 the cost of shipping it the same distance in a 10,000 ton tanker. Building a 300,000 ton tanker costs less than 1/3 the cost per ton as compared with a 20,000 ton tanker. Crew size of the larger ship is only slightly larger than that of the smaller ship.

2) New methods of rapid cargo handling are being introduced as part of several, large scale, integrated, intermodal transportation systems being developed. Standard (8 ft x 8 ft x 20 ft typically) metal containers, with heavy frame structures capable of supporting their weight piled atop one another, have played a key role in such systems, allowing costly U.S. labor (3 to 5 times more than in competing nations) to handle cargo at over 40 times the maximum rate feasible without containers, and allowing ship loading/ unloading at rates in excess of 80 containers per hour.

Other such systems include unitized and palletized cargo, roll-on/roll-off and float-on/float-off concepts discussed below. These systems allow significant increases in the utilization factor of ships which are part of them and provide a revenue earning capability for these ships 3 to 4 times that of a general cargo ship. In addition, increasing use is being made of modern techniques and equipment for shipboard automation to reduce crew size and the portion of operating cost attributable to labor.

The remainder of this section will discuss the suitability of certain ship types for nuclear propulsion. These ship types are presented in order of decreasing suitability as per the author's evaluation. Appendix II presents other orders of suitability as determined by the several dozen knowledgeable persons who responded to a questionnaire regarding nuclear merchantmen.

1. <u>Container Ships</u> --

The container ship is probably the most economically viable ship type for nuclear propulsion today, since it can readily meet all 3 of the necessary basic criteria: high power, long routes, and high utilization; in addition, the high-value cargo it normally carries is generally plentiful enough to ensure full loads round trip. Figure V-1 shows the recent, dramatic increase in SHP of these vessels in response to the desires of certain, high-value cargo shippers of rapid delivery of their goods. To

take advantage of the economies of scale discussed above, the displacements of these vessels have also increased remarkably. Since over 70% of the world's general cargo is susceptible to containerization with its shorter delivery times and reduced pilferage and loss, a large market lies ahead for these ships.

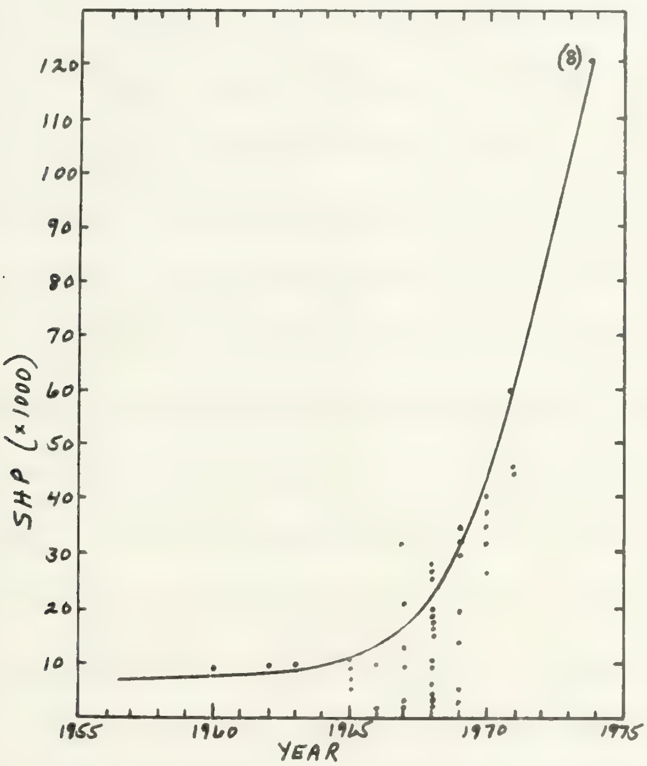

Figure V-1 Trends in Container Ship Horsepower -- Ships in Service or on Order

On a long trade route a conventional container ship requires typically 6,000 tons or more fuel oil unless it is refueled enroute, with consequent delays in cargo delivery. The nuclear container ship, by eliminating the

need for this high bunkering capacity, has greater cargo carrying capacity and therefore greater revenue earning capability. This is especially true for higher value cargo which can command premium shipping rates for fast delivery. Because this high value cargo (typically $1400/ton) normally consumes considerably more of the ship's bale cubic capacity than would the lower value (typically $400/ton) general cargo, container ships tend to be considerably larger than general cargo ships; typical cargo volumes are 100 ft^3/dwt on a container ship and 60-70 ft^3/dwt on a general cargo ship. Needless to say, a high degree of reliability of both the ship and the port turnaround facilities is necessary for this economic viability to be realized. Optimum routes for initial nuclear container ships would be between widely separated, economically advanced areas, since these areas are more likely to have modern, fast turnaround port facilities, high value cargo supplies or needs, and relatively high cost fuel oil.

In addition to the purely economic advantages of a nuclear containership, some important benefits in safety analysis can also be realized. New terminals being developed for container operations are comparatively more remote from population centers than the older cargo piers. In addition, the substantial enclosed area for container receiving and storing is a positive factor in any safety review, affording a ready, controlled access area in the event an accident occurs pierside. Also, the number of ports of call for a

container ship is less than for a general cargo ship because high value cargo tends to concentrate at a limited number of major ports on most trade routes. This reduction in ports of call on long trade routes also decreases round trip time and increases ship utilization.

In the container ship category for potentially economically viable nuclear propulsion in the near future might be included 2 other new ship types:

1) lighter-aboard-ship (LASH), where cargo is stowed aboard covered lighters typically 50 ft long x 20 ft wide; these lighters are hoisted aboard much like containers, or the lighters can be loaded barges brought on board by an elevator at the stern -- the float-on/float-off sea barge concept. The barge carrier becomes more competitive as port congestion increases and as underdeveloped countries, which lack full scale port facilities, increase their needs for overseas shipments.

2) roll on/roll off, with containers loaded topside in stacks, and trailers (loaded with cargo) rolled into the lower decks through ramps at the stern.

2. Passenger Ships --

Large, high speed passenger ships operating on long routes could operate on a schedule that would give high utilization, thereby giving them all the elements needed for economic viability with nuclear propulsion -- except possibly one: fare-paying passengers. Although the ship type is well suited for nuclear propulsion, the future of

large passenger vessels appears uncertain in the face of stiff competition from sub- and super-sonic aircraft promising arrival at the destination just a few hours after departure. If only passengers could be guaranteed for the life of the ship, a very strong case for a nuclear passenger vessel could be made, since the fastest marine passenger service today (other than on the relatively short Atlantic routes) is about 20 knots. The reason for this slow speed on the longer routes is the large bunkers required for high speed. For example, the liner UNITED STATES requires 5,000 tons of fuel oil for an Atlantic crossing at 35 knots; this ship is 990 ft long, 47,250 tons displacement 240,000 SHP, has 4 screws and burns over 50 tons of fuel per hour at 35 knots. Nuclear propulsion could, of course, provide high speeds without the excessive bunkering required for conventional liners.

3. Icebreakers --

Several reasons combine to make icebreakers quite suitable vessels for nuclear propulsion, although without a steady source of commercial revenue it is somewhat futile to try to convincingly argue that a nuclear icebreaker is economically competitive. Since they operate in remote areas for extended periods, the range of operations of a conventional icebreaker is often jeopardized by the threat of fuel exhaustion. Fear of being trapped in the ice with fuel depleted decreases the present length of seasons for polar expeditions and often dictates abandonment of a

scientific effort. A nuclear icebreaker, if beset in the ice, would have no shortage of fuel and could be on the line fully operational as soon as ice conditions permitted.

Since the Arctic constitutes a vulnerable flank to North America and the natural resources potential of the area has attracted world interest, nuclear propulsion could also have a strategic application in this area. Assistance to downed aircraft and submarines by nuclear icebreakers might be possible in seasons now considered infeasible for conventional vessels outside the usual 3-3 1/2 months of summer operation (Utilization factor for conventional icebreakers is very low, typically 15-20%). Nuclear power would also permit installation of increased SHP to improve icebreaking capability and would free the vessel from the requirement for frequent bunkering enroute. Fuel costs for icebreakers tend to average 50% more than for other ships, due to their remote area of operation. The utility of nuclear icebreakers is attested by the U.S.S.R.'s building follow-on vessels after operating the LENIN extensively in ice.

4. Novel Ship Types --

Three types of novel ships are considered here. In spite of the suitability of these ship types for nuclear propulsion, it will probably be many years before these ship types will come into commercial service. Similar to nuclear propulsion itself, the major obstacle to progress in the application of new ship types is monetary; the cost

of research and development and the financial risk entailed by use of unproven designs are large hurdles that tend to be overcome only by generous government funding or a high degree of assurance of financial reward in the form of net profit.

a. Submarine Cargo Carriers --

Many versions of submarine cargo carriers are possible, carrying all types of cargo from oil to containers. This ship is ideally suited for nuclear propulsion in order to free its power plant from dependence on the atmosphere. The ship is also best suited competitively for routes that are generally ice-covered. A cargo submarine could have low outfitting costs due to elimination of the need to batten down cargo to withstand storm-induced forces. A roll-on/roll-off concept could be used so that, for example, 4 trains of railroad cars could be carried on 2 deck levels. Because of freedom from storm-induced slowdowns to preclude cargo or ship damage, the ship could promise on-schedule delivery of cargo with a higher chance of success than a surface ship. Three elements would be needed to make a transportation system work:

1) the ship itself -- for example (ref's 116 and 118), a 38 ft OD, 1,400 ft long, 200,000 SHP, 2 screw, 40 knot, 5,000 dwt submarine designed for 200 ft operating depth.

2) special port facilities for (possibly automatic) docking, loading and unloading of the ship, and

3) a transoceanic guidance system such as a

sea-floor cable with transponders along the route(s), or inertial guidance equipment aboard the ship.

b. Semi-Submerged Catamaran --

This ship consists of a flat, horizontal box girder structure (the upper, dry hull) supported by 2 near-surface, submerged, parallel, horizontal-axis bodies of revolution via 2 sets of streamlined struts. It is adaptable to efficient roll-on/roll-off service and has vastly superior motion performance compared to an ordinary displacement ship in a rough seaway due to its small water-plane area. Above 24 knots the SHP requirement is less than that for an ordinary, high speed, displacement hull. Such a ship could benefit greatly from a non-air breathing power plant with constant fuel weight independent of voyage length.

c. Composite Ships --

This concept is analogous to the familiar tractor-trailer rig on the highways today and is somewhat of an extension of the tug-barge arrangement used extensively on inland waterways. It consists of a powered unit latched to and pushing an unpowered cargo unit; both units together have the usual sea-going displacement hull. If a relatively large number of cargo units were available, the power unit could conceivably attain utilizations in the neighborhood of 95-98%. By providing harbor-mouth transfer of the cargo unit to a port tug, the power unit could even eliminate port entries and remain remote from large population areas except

for refueling and maintenance periods, thereby enhancing its performance from a nuclear safety standpoint. Some design problems must be solved to make this concept totally workable, however. These mostly involve the design of the juncture between the 2 units so it will both be readily coupled and decoupled and be capable of withstanding the forces and moments induced by a rough seaway; also included are considerations of matching the draft and the trim of the 2 units, and of stability and controllability of the power unit when operating by itself. Of course, it goes without saying that for nuclear economic competitiveness such a concept would have to be associated with high power and long trade routes in addition to high utilization.

5. Supertankers --

Although their shaft horsepowers are not as high as for containerships, recent trends are for higher and higher SHP's with current ships approaching 50,000 SHP. Fossil fuel costs for these ships are usually lower than for other ship types due to bunkering in oil-producing areas. By 1969, 200,000 ton deadweight tankers had become commonplace; 300,000dwt vessels are now in service and vessels up to 1,000,000 dwt are being designed. The rationale for these large sizes is simply economics:

- ship weight and ship hull steel cost vary as ship size to the 0.9 power

- ship construction labor cost varies as ship size to the 0.6 power

- power plant cost varies as SHP to the 0.6 power
- ship operating crew varies as ship size to the 0.2 power
- fuel operating costs for constant speed vary as ship capacity to the 0.7 power

A size of 600,000 dwt, however, appears presently to be about the size where problems of draft and maneuverability tend to offset the gains of further economic savings by increasing ship size. For example, a current, 300,000 dwt, 16 knot tanker takes a good 2 miles to stop using maximum power backing down. Also, ships with drafts over 60 to 80 ft have difficulty transiting such major areas as the Strait of Dover, the Malacca Straits, the gaps in the Indonesian Islands and the Sunda Strait; nothing foreseeable indicates this limitation will be overcome in the next 40 to 60 years.

In spite of the possible economic viability of nuclear supertankers in the distant future if SHP's continue to increase as petroleum becomes more in demand and less in supply, problems of reactor plant safety may preclude the use of nuclear propulsion for ships of this type. For example, such features as the low maneuverability of such a large ship and the relatively high explosion and fire hazards associated with carrying vast quantities of oil have in the past caused concern among statutory bodies such as the AEC and the Advisory Committee on Reactor Safeguards when nuclear

powered tankers were proposed. Without a significant increase in the demand for and the price of oil, however, it is difficult to foresee SHP increase to the point of economically viable nuclear propulsion in these ships in the future.

6. <u>General Cargo, Other Tankers and Bulk Carriers</u>--

At the present time, neither grains, ores and other bulk cargoes nor general cargoes tie up large sums of money at the producer's end while they are in transit, so these ships tend to be relatively slow. Even the large (up to 200,000 dwt) bulk-ore and bulk ore-oil carriers currently being planned have SHP's roughly equivalent to tankers of this size. In addition, the weight of machinery and fuel in the bulk carriers is generally low in comparison with cargo weight such that the few extra tons of cargo carrying capacity afforded by nuclear propulsion do not appear worth the extra initial cost. In addition, these ships often carry cargo in one direction only and make the return trip ballasted with sea water, greatly reducing their revenue earning capacity. In-port turnabout times tend to be long and transit speeds low. All of these reasons make it difficult to foresee any economic advantages for nuclear propulsion in these ships for at least the near future. Other ship types included in this category are those designed for just one type of cargo, such as liquified natural gas (LNG), liquidied petroleum gas (LPG), liquid sulphur, fish

flour, and iron ore slurry. A further impediment to the use of nuclear propulsion for some of these ships is the fire or explosive hazard associated with their cargoes.

7. Hydrofoil and Hovercraft --

In general, both hydrofoil and hovercraft vehicles require high SHP, low specific weight power plants. NASA research indicates that technical feasibility for use of water moderated and cooled reactors is currently limited to hovercraft above 10,000 tons; the largest current hydrofoils and hovercraft vehicles weigh only hundreds of tons. There is serious doubt as to whether or not these vessels will ever be built weighing thousands of tons, but a detailed design for a gas cooled direct cycle reactor plant for a 4,000 ton surface effect ship has been developed (ref 113). This plant is based on advanced technology which currently does not exist, such as:

- 2,000 psia, 1550F reactor outlet helium conditions
- heavy metal hydride shield materials
- core volume one-seventh that of today's shipboard PWR's for the same power output
- ultra light weight electric drive based on 4,000 Hz. systems

C. FORECAST OF FUTURE APPLICATIONS--

As stated in Section IV, one cannot make a meaningful statement to the effect that nuclear propulsion either is or is not economically competitive today. Each ship type

and each trade route has its own distinctive characteristics which can strongly influence economic comparisons. Published literature indicates that a second generation of nuclear merchantmen might be economically competitive if the "right" type ship and the "right" route are selected, and that a third generation of nuclear ships would surely be competitive.

The "right" ship type for a second generation of nuclear merchantmen appears to be the high-speed container ship. The demand for these ships carrying high value cargo at premium rates has been established. Short turnaround port facilities utilizing rapid cargo handling equipment have been built in many major ports and are being built or planned in many others. A fleet of 3, 4, or more of these ships operating on a long trade route would provide the experience needed to determine where additional development effort should be applied in order to improve the competitiveness of these plants. These ships would be the steppingstones to truly competitive nuclear merchant vessels, just as Yankee and Dresden were the steppingstones to truly competitive nuclear central stations. The surest way to reduce costs and widen the range of commercial application of nuclear propulsion is to accumulate construction and operating experience with nuclear ships.

The largest single issue still looming over the decision to build a second generation of nuclear ships is financing. The solution to this issue, however, ideally should be

relatively straightforward. He who contributes the most toward the development of a nuclear merchant fleet should be, ideally, he who stands to gain the most from its development.

The shipowner certainly stands to benefit from a fleet of nuclear ships, especially in a climate of increasing fuel oil prices and relatively stable nuclear fuel prices. The shipowner, then, should be willing to pay at least as much of the construction cost as he would have to pay if the fleet were conventional. Equitably balancing the inevitable risks associated with a new machinery plant, the longer nuclear ship construction period (5 years vs. 2 1/2), and certain higher nuclear costs (for insurance, for special shore staff for compliance, port entry and refueling work, and for maintenance and repair) against the potential fuel savings gains, he should be willing also to apply to apply any profits he might make above a certain level (say, 15% ?) to the development of future nuclear propulsion plants.

The nuclear industry also stands to benefit from a fleet of nuclear ships, although the benefit to this industry, with its present (and sure to continue) burgeoning nuclear central station business (over 100 large commercial water reactor systems are under construction or on order with large increases projected for the near future), might be less than that to the shipowner, at least until large fleets of nuclear merchantmen are on order. This industry, then, should at least be willing to apply significant amounts of its resources to the develop-

ment of cheaper, more reliable nuclear plants based upon initial difficulties and experiences in the construction and operation of the second and subsequent generation nuclear ships.

The largest potential benefit, however, from the commercial maritime application of nuclear propulsion in the United States appears to be to the U.S. citizen, for a number of reasons. Some of these are as follows:

1) Nuclear propulsion may be the only remaining way to bolster the declining U.S. merchant marine and regain for it a competitive position in the worldwide shipping industry. A strong and competitive merchant marine is the only practical way for the United States to curb its increasing dependency on other countries for the transport of its oceanborne trade and for its supply of fuel oil.

2) Nuclear propulsion would improve the nation's balance of payments problem: in the near term, by eliminating the purchase from foreign sources of $50 to $100 million of fuel oil per ship over its lifetime; and in the long term, by affording an export sales opportunity of $5 billion of high powered nuclear ship propulsion equipment by 1990 if only 10% of the projected worldwide market for this equipment were secured. Supplying core reload fuel for these nuclear plants would further aid the balance of payments. If this market is to be secured, however, it is important that the U.S. demonstrate the reliability of its high powered nuclear propulsion plants by operation of a nuclear merchant fleet before the lead

in this field is taken by other countries such as W. Germany, Japan and the U.S.S.R. The balance of payments would also be improved by a competitive U.S. merchant marine by an increase in the percentage of U.S.-foreign trade shipping costs paid to U.S. shipowners and a decrease in that paid to foreign shipowners.

3) Nuclear propulsion would also provide other, less tangible, less quantifiable benefits, such as enhanced national prestige, enhanced national security, reduced environmental infringement, and potential spinoff benefits to other industries (an analogy with the space program could be made). Today, more than ever before, political influence and trading interests are generated by an economic presence in an area; this is especially true in newly developed countries. U.S. influence in such countries would be greatly enhanced by a viable merchant marine. In addition, the defense posture of the U.S. would be enhanced by the availability of a fleet of high speed ships which could support its mobilization requirements independent of a tenuous fossil fuel supply in an emergency. Future national conflicts may not afford the opportunity, even on a crash basis, of remedying a neglected, decadent peacetime merchant marine.

Because of the large benefit of a strong, competitive merchant marine with fleets of high-powered nuclear ships, the United States as a country should be willing to provide the ship construction subsidy and other incentive support

needed to get the first fleet of nuclear ships built, even if the second generation propulsion plant utilized in these ships were not fully economically competitive. Since the cost of capital has such a strong influence on the economics of nuclear propulsion (0.05-0.1 mills/SHP hr per 1% change in interest rate), a guarantee of low interest rate capital might be another appropriate incentive. An extension of the special liability insurance coverage set up for SAVANNAH might also be appropriate. It is not unreasonable to assume that some or all of this incentive support will be compensated for by future increased revenue from taxes paid by U.S. shipowners and nuclear industry and by the reduction in gold outflow associated with the nation's balance of payments deficit. Whatever the incentives used, however, a <u>positive attitude</u> of <u>cooperation</u>, rather than the seemingly internecine squabbling, should be able to produce an equitable financial arrangement between the shipowner, the nuclear industry and the government.

To obtain any of these benefits, though, it is necessary to build and operate nuclear ships. To build follow-on nuclear merchant ships with their higher capital cost, it is necessary to have a clear-cut government policy, or at the very least a government consensus that such a project is in the best national interests so that priorities and budgets can be appropriately adjusted. Without such a policy or consensus, a fleet of U.S. nuclear merchantmen and a strong

and competitive U.S. merchant marine will remain only an illusory vision of farsighted men. Without this policy or consensus, none of the 500 merchant ships with over 100,000 SHP that current MarAd projections indicate will be built by 1990 will be propelled by a U.S. built nuclear plant. Because of the enhanced defense mobilization capabilities afforded by both nuclear merchantmen and a strong U.S. merchant marine, the solid support and encouragement of the Navy Department and the Department of Defense would be helpful; such support has not been given in the past.

Just where this needed consensus stands today is not certain, but the only existing government program, geared for near term construction of second generation commercial nuclear ships, should find out within the next year. During the time AEC interest in merchant ship nuclear propulsion was waning after 1966, MarAd began an orderly, step-by-step program directed toward building the first U.S. fleet of commercial nuclear ships in mid-1973. Babcock & Wilcox was contracted to design the propulsion plant, subcontracting George G. Sharp Co. and a major shipyard to perform ship-related naval architectural and pre-construction engineering work.

A Preliminary Safety Analysis Report for this design was submitted in 1968, on which the AEC provided comments for future program guidance. Detailed design work and economic analyses evolved into detailed pre-construction

engineering. As an estimate of the present state of economic competitiveness of this design, Figure V-2 presents some results of this detailed analysis. Table V-3 below lists the main assumptions which went into this analysis. The capital cost difference between nuclear and conventional ships in Figure V-2b peaks at about $19 million because of 1) improved economics of construction with increased reactor size and 2) the cost of multiplicity for the fossil boilers (maximum size is about 60,000 SHP each). Figure V-2d shows the strong dependence of nuclear economic competitiveness on power level and interest rate.

Table V-3 Assumptions for B & W/MarAd Economic Analysis of CNSG Application

- Container Ship
- 2,216 containers at 80% load
- 120,000 SHP Nuclear/128,000 SHP Conventional
- Beam - 105.5 ft
- Ship Life - 25 years
- Service Speed - 30.9 knots
- Round Trip Distance - 21,550 miles
- Ship Availability - 354 days/year
- Cost of Capital (interest rate) - 12%
- Construction Subsidy - 45%

Another part of this economic analysis compares a 5 ship fleet of 24 knot, 120,000 SHP nuclear propelled tankers and an 8 ship fleet of 15.5 knot, 35,000 SHP

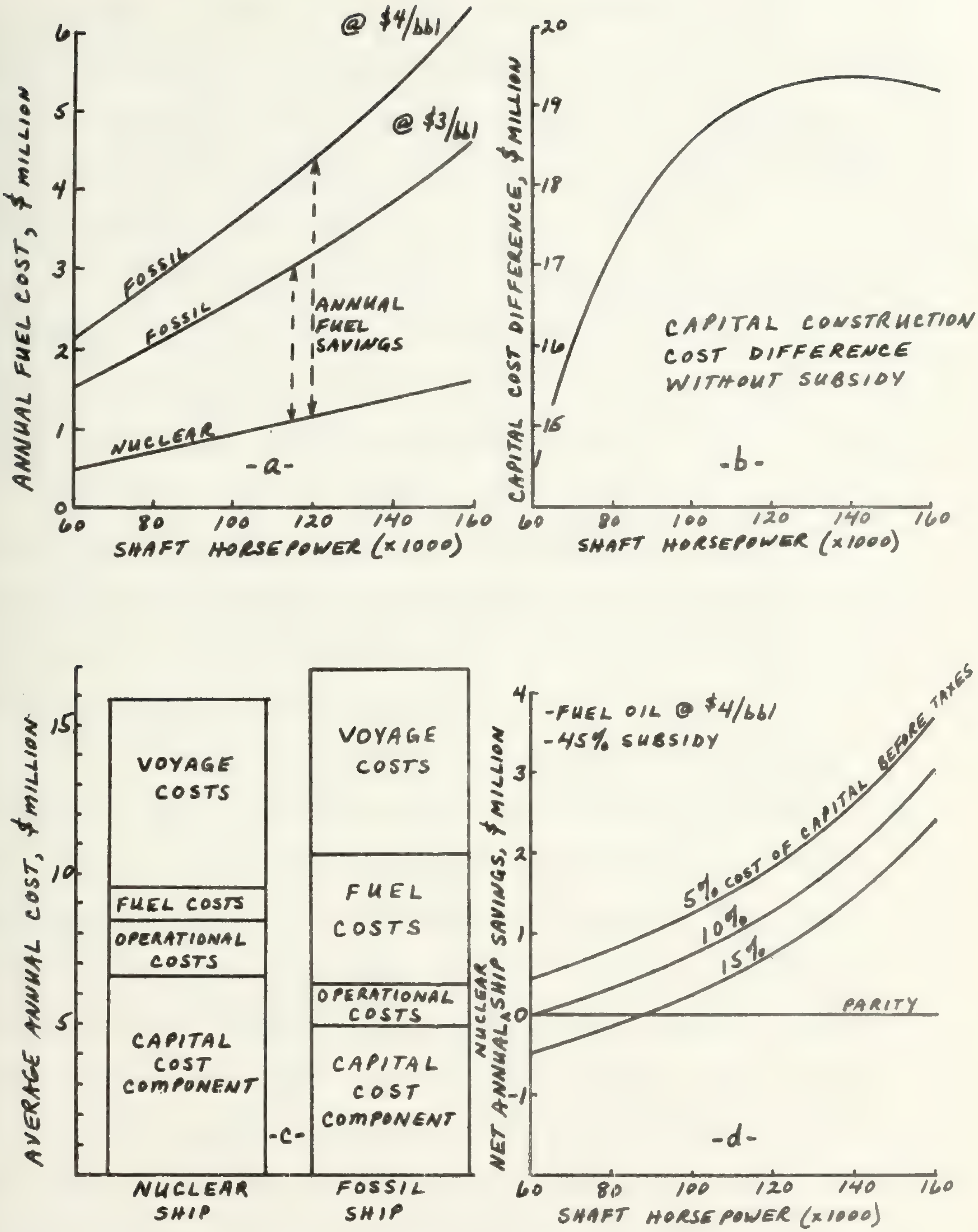

Figure V-2 Results of B&W/MarAd Economic Analysis of CNSG Plant with Assumptions of Table V-3

conventional tankers. All ships are 250,000 dwt and the 2 fleets carry the same amount of oil per year. Capital costs of the 2 fleets are the same, while the net earnings per year of the nuclear fleet is $5-6 million more than those of the conventional fleet.

A revised Preliminary Safety Analysis Report will be submitted to the AEC by the end of 1972 as the basis for reactor construction permits and licensing for these plants. A detailed manufacturing cost and ship construction schedule will be prepared by early 1973. The goal of this program is to place 3 nuclear-powered merchant ship contracts in mid-1973 while concurrently satisfying the requirements of the various regulatory agencies involved at various stages of design and construction.

To this end, B & W and MarAd are now actively engaged in detailed discussions with several shipowners, as well as shipyards, naval architects and other segments of the marine community. Shipowners and operators have been given large amounts of data so that they can evaluate for themselves the economic competitiveness of nuclear power in their ships, operating in their fleets and to their ports of call. In these discussions with shipowners and operators, the areas of most expressed concern have not been technical feasibility or even the general economic competitiveness of nuclear propulsion. The questions of most concern have been related to complex government administrative matters, such

as insurance, domestic and foreign licensing, and ship construction subsidy. In these matters, especially in the matter of ship construction subsidy, it is up to the government, both the legislative and the executive branches, to provide the leadership and the direction without which nuclear merchant fleets will not be built.

It would be a mistake to overlook the economic realities of getting a nuclear merchant fleet started. It is an expensive project that must take its rightful place in the listing of national priorities. But it would also be a mistake to overlook the economic and strategic realities of allowing the decline of the U.S. merchant marine to continue while the one remaining key to its renaissance is at hand and ready for use.

The author is guardedly optimistic that the cautious and deliberate approach of the Maritime Administration will lead, with needed government consensus and support, to construction in 1973/1974 of the small beginning of a U.S. nuclear maritime fleet, and that with cost-reducing improvements resulting from experience with these second generation plants this nuclear merchant fleet will be added to in large numbers in the 1980's and beyond. The author sincerely hopes his is not just an illusory vision. One last point: With limited research and development funds available in the near future for maritime nuclear propulsion, maximum economic benefit can probably be realized by concentrating on adapta-

tion and improvement of successful nuclear central station technology; the advent of reactor types other than water-cooled and -moderated for maritime applications should probably await the successful development and optimization of these reactor types for central station or other applications.

VI. CONCLUSIONS

The main conclusions of this study of marine nuclear propulsion are as follows:

1. Nuclear propulsion has for many years been technically feasible in practically any ship type operating on practically any trade route.

2. There are many advantages to be gained from nuclear marine propulsion. For naval ships, these advantages are primarily tactical and include high speed and long endurance independent of the atmosphere and tenuous fuel oil supplies. For commercial ships, the advantage is primarily economic: the promise of higher net earnings.

3. Economic competitiveness of nuclear propulsion has not been achieved to date. Nuclear ship operating characteristics presently necessary to achieve economic competitiveness include: high power, long routes, and high ship utilization.

4. The ship type presently most capable of achieving economic viability with nuclear propulsion appears to be the high-speed container ship operating on long trade routes between technologically advanced countries. These ships are currently an integral part of new, intermodal transportation systems designed for fast delivery of high value cargo at premium shipping rates. Since at least 70% of the world's oceanborne general cargo is susceptible to containerization, with its reduced pilferage and loss and its

easier handling and faster delivery, the future demand for these ships seems certain. Japan and W. Germany recently announced firm plans to begin production of a fleet of nuclear container ships in the near future.

5. The strength and economic competitiveness of the U.S. merchant marine has been following an historically recurrent pattern of steady decline since the end of World War II. The U.S. merchant fleet is now decadent and virtually non-competitive in world maritime shipping, so that the United States has become heavily dependent on the ships of other nations to carry its oceanborne trade. This situation has significant adverse effects on the nation's balance of payments deficit. The primary reason for the non-competitiveness of the U.S. merchant fleet is high costs, mostly those of U.S. labor. The subsidy system set up by the Merchant Marine Act of 1936 has not been effective in checking the current decline of the U.S. merchant marine.

6. Nuclear propulsion may be the last remaining way to restore the U.S. merchant marine to a position of economic competitiveness in the maritime shipping industry. To date, lack of a clear-cut government policy toward follow-on nuclear merchantmen, and indeed toward the U.S. merchant marine peril in general, has prevented the large government investment that appears necessary to construct a small fleet of follow-on nuclear merchant ships using available, second generation propulsion plants. There is a

good possibility that a third generation and, more certainly, subsequent nuclear propelled ships will be economically competitive without large amounts of government aid. The United States stands to gain considerable strategic and economic benefits from having a viable nuclear maritime fleet in its merchant marine.

7. The author is guardedly optimistic that the Maritime Administration's current, thorough and deliberate program leading to construction of a small fleet of second generation nuclear merchantmen in 1973/74 will be successful. The success of this program hinges primarily on the willingness of government leaders in both the Legislative and the Executive Branches to assign to the program a national priority commensurate with the level of government funding that will be necessary.

8. Achievement of nuclear marine economic competitiveness is hindered primarily by 2 problems stemming from the fundamental nature of the fission process whence nuclear energy comes. These 2 problems are: 1) protection of the crew from the intense radiation given off during the fission process, and 2) protection of both the crew and the general public from potential release of the prodigiously radioactive fission products both during normal operation and in the event of any foreseeable, credible accident. Although these problems make achievement of a low weight, low volume, low cost nuclear propulsion plant very difficult, experience with

construction and operation of follow-on nuclear ships should indicate areas wherein economies can be achieved with some limited amount of additional design or development work.

9. All nuclear propelled ships operating today are based on pressurized water reactor technology. Although design studies of nuclear propulsion plants utilizing other reactor types may indicate certain distinct advantages as compared with the PWR plant, there also exist disadvantages which, at least for the present, may in practical application to shipboard propulsion outweigh the apparent advantages. For at least the near future, it would seem prudent to continue building and operating nuclear marine propulsion plants based on technology which has been extensively proven by central station or other operating experience and which can be satisfactorily adapted for shipboard propulsion. In this way, maximum near term economic gains can be realized without unwarranted research and development expenses.

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APPENDIX I

DETAILS OF SPECIFIC NUCLEAR MARINE PLANTS

A. N.S. SAVANNAH (ref's. 23 through 27)

1. GENERAL --

SAVANNAH is a single-screw, 9 compartment, combination general cargo and passenger ship of 10,000 tons deadweight, with sheltered deck, raked bow, and modified cruiser stern. Slightly larger than the Mariner class, SAVANNAH has a full load displacement of 21,850 long tons at a mean draft of 29 1/2 ft; design speed is 20 1/4 knots at 20,000 shaft horsepower. Other principal characteristics are:

Length Over All	595 ft 6 in.
Beam, Molded	78 ft 0 in.
Draft, Light Ship Condition	18 ft 6 in.
Displacement, Light Ship	11,850 long tons
Cargo Deadweight/Bale Cubic	9,250 l.t./746,200 cu ft
Passengers (one class only)	60
Officers, crew and others	124
Standard of Subdivision	2 compartment

SAVANNAH'S cargo handling equipment was the lightest developed before 1960 for the modified Ebel rig and is fitted for very rapid handling of cargo. Hydraulically operated anti-roll stabilizer fins are mounted on port and starboard sides amidships. Safety and reliability were emphasized throughout the SAVANNAH power plant design, so that the only vital units without some form of installed backup are the rudder, propeller and shafting.

2. SHIP ARRANGEMENT AND STRUCTURE --

The general arrangement of SAVANNAH is shown in Figure A-1. The hull is built on a transverse framing system except in the innerbottom, which is a combination of transverse and longitudinal framing especially stiffened below the Reactor Space to withstand all anticipated loadings, including grounding. The decks outboard of the Reactor Space have been specially strengthened to form the principal barrier for collision protection. Backup barriers providing additional collision resistance are formed by a complex consisting of longitudinal bulkheads, a 35 ft longitudinal, laminated collision mat 24 in. thick, made up of 1 in. steel and 3 in. redwood sheets, on each side of the containment vessel, and the concrete secondary shield. The containment vessel rests on a foundation consisting of six longitudinal girders integrated with deep transverse saddles. Secured to the foundation at the aft end only, the vessel is free to expand and contract both laterally and longitudinally, yet is restrained from gross motion by the collision mats and chocks at C-deck level.

The ship's power plant is located in two compartments. The Reactor Space, located amidships, houses the various components of the single reactor system, most of which are inside the containment vessel. The Machinery Space, immediately aft of the Reactor Space, houses the propulsion system and the central control room from which the entire power plant is operated.

3. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

a. CONTAINMENT VESSEL --

The arrangement of the principal reactor system components in the containment vessel is shown in Figure A-2. The SA212B Carbon steel containment vessel is a horizontal cylinder with hemispherical ends, 35 ft in diameter and 50 1/2 ft long; it is surmounted by a centrally-mounted cupola 13 1/2 ft in diameter and 16 1/2 ft high, which encloses the control rod drive mechanisms and has a full diameter hatch for maintenance and refueling. Two other, 3 1/2 ft diameter, access hatches and two 24 x 18 in. manways are provided in the vessel for maintenance. Including these hatches and manways, there is a total of 85 penetrations in the containment vessel for piping, access and the over 300 electrical cables needed. Most of these penetrations are in the lower half (concrete shielded) of the vessel so that additional, compensating shielding is not required.

The shell thickness of the vessel ranges from 2 3/8 to 4 inches, enough to withstand the design maximum credible accident: a primary boundary rupture, resulting in the flashing of all primary coolant to steam, with an associated internal pressure of 186 psi and an increase in temperature of 300F. Containment vessel integrity in the event of sinkage in deep water is ensured by automatic flooding valves which open at an external head pressure of 100 ft of water to admit sea water and prevent rupture, and reclose when internal and external pressures have equalized.

b. RADIATION SHIELDING --

The 418 short ton primary radiation shield, designed to permit entry into the containment vessel for maintenance within 30 min after the reactor is shut down (radiation level less than 200 mrem/hr), consists of a 33 in. annulus of water in a 17 ft high tank surrounding the cylindrical portion of the reactor pressure vessel, with an outer layer of about 4 in. of lead. The 2000 short ton secondary shield, shown in Figure A-3, consists of two parts: 1) a reinforced ilmenite-aggregate concrete skirt 3 to 4 ft thick surrounds the lower half of the containment vessel; and 2) lead slabs 2 1/2 to 6 in. thick, overlaid with 11 1/2 to 8 in. of polyethylene, are mounted as a continuous 14 in. thick shield directly on the upper half of the vessel and on the cupola.

The concrete shield is extended forward to form a rectangular "undershield vestibule" which houses the systems for primary coolant purification, gaseous waste collection, and drain and waste collection. Three-eighths inch construction gaps between adjacent lead sheets are filled by tightly caulked lead wool rope to prevent radiation streaming, while non-overlapping gaps are left between adjacent polyethylene sheets to allow for its high coefficient of thermal expansion. The 3 ft x 3 ft lead slabs are held in place by 1 1/2 in. diameter steel studs welded to the containment vessel on 18 in. centers. Polyethylene sheets are 8 ft x 4 ft x 1 in. thick and are held in place by 1/2 in. diameter studs threaded into

the lead-holding studs and by 2 inch aluminum spiral nails driven into preceding layers and underlying lead slabs; edge gaps are 1 1/4 ± 1/8 in. for the 8 ft length and 5/8 ± 1/8 in. for the 4 ft breadth. Maximum design dose rate outside the secondary shield is 5 rems/year (see Figure A-3). The entire reactor system, including containment vessel and supports and all shielding, is approximately equal in weight to the bunker oil capacity of a Mariner class ship.

c. PRIMARY SYSTEM --

The primary system consists of the reactor and two parallel,symmetrically arranged, 16 1/4 in. OD coolant loops, each containing:one horizontal, U-tube, U-shell boiler; two 5,000 gpm, 2-speed, 70 psid,centrifugal, "canned-motor" pumps; two pump outlet check valves to prevent backflow with only 1 pump operating (small holes are in the valve discs, however, to allow temperature-equalizing backflow); and two electric motor-driven, loop isolation, gate valves. The system is schematically illustrated in Figure A-4. At maximum reactor power of 70 MW, the primary coolant temperature rise from reactor inlet to outlet is 23.4F at an average temperature of 508F. Boiling of the light water primary coolant/ moderator is prevented by maintaining a pressure of 1,735 psig on the coolant; this is accomplished by the pressurizer, a 154 cu ft cylindrical pressure vessel containing primary coolant and a 92 cu ft steam bubble maintained by 160 electric heaters of 222 KW capacity and an internal spray system.

Heated coolant passing through the 3/4 in. OD x 0.072 in. minimum wall thickness, 304 stainless steel boiler tubes generates steam at pressures from 730 psia at no load to 472 psia at full load; maximum load steam flow is 265,850 lb/hr at a steam quality in excess of 99.75%. The shell side of each steam generator is connected to an upper drum by 13 risers and 8 downcomers designed to ensure boiler circulation at all power levels and ship attitudes. Cyclone separators and scrubbers in the upper drum remove excess moisture from the steam. Primary system components are designed for 2,000 psia, secondary system (boiler shells, steam drums, and piping) for 800 psia. Activated corrosion product inventory in the primary system is minimized by fabrication of all surfaces in contact with the coolant using austenitic stainless steel, ASTM A376-TP304, and by use of a coolant purification system; this system consists of a demineralizer (plus 2 installed spares) containing 17.5 cu ft of mixed cation-anion type resins through which is circulated 20 gpm of primary coolant at 110F, 40 psia.

d. AUXILIARY SYSTEMS --

In addition to the pressurizing and purification systems, other systems associated with the primary system are necessary for safe and reliable reactor plant operation. These are:

i) Pressure Relief System -- prevents pressure at any point in the primary system from exceeding 2,000 psia, and condenses and contains primary relief valve effluent in

a 1,000 gallon tank containing 450 gallons of quenching water; prevents secondary pressure from exceeding 800 psia.

ii) Hydrogen Addition System -- effects recombination of oxygen produced by radiolytic dissociation of coolant in the core, by maintaining a hydrogen concentration of 20-40 cc/liter (at standard temperature and pressure) in the primary coolant.

iii) Buffer Seal System -- returns purified coolant to the primary system via the control rod drive mechanism buffer seals, thereby preventing primary coolant leakage where control rod drive shafts penetrate the reactor pressure vessel head; also automatically provides primary system makeup water from an 87 cu ft surge tank via 3 parallel charging pumps to maintain proper pressurizer water level.

iv) Emergency Cooling System -- automatically provides 200 gpm primary coolant flow through the core and through a sea water-cooled heat exchanger upon loss of normal electrical power to primary coolant pumps, thereby preventing the core from overheating.

v) Soluble Poison Addition System -- provides an alternate (emergency) means of manual reactivity control by deliberate addition of boric acid (H_3BO_3) to the primary system using a 24 gallon mixing tank and a 1/2 gpm addition pump.

vi) Sampling System -- provides periodic and continuous primary coolant samples for analysis of chemistry and radioactivity levels, including fission product concentra-

tion measurement to detect possible fuel cladding failure.

vii) Intermediate Cooling System -- provides 95F fresh water cooling flow for various reactor system components such as primary coolant pump windings, letdown coolers upstream of purification system demineralizers, neutron shield tank, control rod drive hydraulic power supply, and air conditioners; the fresh water cooling flow is in turn cooled by sea water flow in 2 parallel fresh water-sea water heat exchangers.

viii) Containment Vessel Air Conditioning System -- maintains the atmosphere inside the containment vessel below 130F and 72% humidity during reactor operation, thereby preventing untimely failure of the electrical insulation on the large amount of wiring in the vessel.

ix) Gaseous Waste Collection System -- concentrates and contains gaseous radioactive wastes when the ship is operating in confined waters or under weather conditions not favorable for controlled release of such wastes to the atmosphere; adsorbs fission product gases on charcoal maintained by liquid nitrogen cooling at -280F. Associated systems collect radioactive liquids from heatup (2,400 gallons), sampling, and other sources and store them in the 10,000 gallon capacity tankage provided for this purpose, for eventual discharge to a disposal facility or to the sea when the ship is far from land.

e. THE REACTOR --

i) The Core --

The reactor core makes up essentially a right circular cylinder whose active region is 62 in. in diameter and 66 in. high. It is composed of 5,248 vertical fuel pins assembled in rectangular array in 32, 8 1/2 in. square fuel elements, as shown in Figure A-5. The fuel pins, centered 0.663 in. apart in a square lattice, are 0.035 in. thick, 0.50 in. OD, TP304 stainless steel tubes which encase compressed and sintered (mean density: 10g/cc) uranium dioxide pellets, each 0.4245 in. in diameter and 0.50 in. high. Non-fuel spaces inside these pins are filled with helium to enhance heat transfer. Fuel pins are held rigidly spaced in the fuel element by 1 in. long stainless steel ferrules brazed between them every 8 in.

Radial power density is flattened by differential fuel enrichment (4.60% U-235 in the outer 16 fuel elements; 4.20% in the inner 16) and by the coolant flow path: coolant enters the reactor vessel near the bottom, flows upward through annular passages on each side of the stainless steel outer thermal shield, and then flows downward through the outer fuel elements; the heated coolant, less efficient as a moderator, is then redirected upward through the inner fuel elements, as shown in Figure A-6. This multi-pass flow path also serves to: 1) permit improved flow utilization without the need for flow-distribution orifices, and 2) reduce, to roughly half, the required total flow rate for satisfactory

heat transfer, thereby also reducing the required coolant pumping power. Fuel conversion ratio for this core is 0.4.

Reactivity control is effected by 21 cruciform-shaped control rods, arranged on a 9.7 in. square pitch, each actuated by an independent drive mechanism mounted above the reactor vessel top head. Each control rod consists of a 62 in. long, neutron-absorbing section with 8 in. tip-to-tip blades 0.375 in. thick; blades are a 0.188 in. matrix of 1.5 weight percent boron (92% enriched in B-10) in stainless steel, clad with 0.094 in. type 304 stainless steel. Attached to this section are an upper, 23 in. long, stainless steel extension and a lower, 59 in. long Zircaloy-2 follower section, the latter to minimize thermal neutron flux peaking in the channel of a raised control rod.

Other reactor characteristics are:

Core heat transfer area	3,778 sq ft
Core metal/water ratio	0.76
U-235/U238/UO_2 loading	312.4/6,787.5/8,200 kg
Average fuel burnup	7,352 MWD/metric ton of U
Average thermal neutron flux, normal power	$7.2 \times 10^{12} n/cm^2$-sec
Maximum/Normal power output	70/64.7 MWt
Coolant flow velocity, outer/inner fuel elements	10/9 ft/sec
Design power distribution factors, maximum to average: radial/axial/local/product	2.0/1.5/1.25/3.75

Average heat flux at

maximum power — 63,500 Btu/hr-ft^2

Max. hot channel heat

flux at max. power — 277,000 Btu/hr-ft^2

Maximum hot channel temperatures:

coolant/pin surface/fuel — 541/623/3,794 F

Core life expectancy — 42,000 MWD (about 3.5 years at normal power 60% of the time and at port power 40%)

ii) The Reactor Pressure Vessel --

Constructed of ASTM A-212 Grade B carbon steel, the 27 ft high and 8 ft 2 in. ID reactor vessel is a vertical cylinder with hemispherical ends; the upper end is bolted on with 48-5 in. studs and can be removed for core loading and unloading. Its 6 1/2 in. thick cylindrical walls and 6 1/4 in. hemispherical walls are internally clad with a 0.11 in. layer of type 304 stainless steel to minimize vessel corrosion and concentration of corrosion products in the coolant. The vessel is protected from excessive thermal stresses due to gamma radiation heating, and from neutron radiation damage, by the three concentric, stainless steel thermal shields shown in Figure A-5.

f. NUCLEAR INSTRUMENTATION --

Ten sets of neutron flux measuring instrumentation cover the entire flux range of the reactor from source level to 150% of maximum power in 3 separate, overlapping channels. These channels provide flux level and rate of

change of flux level signals needed for reactor control and safety. The multiple sets of instruments provided in each channel are operated in a coincidence arrangement such that at least 2 sets of instruments in a channel must concur in indicating an unsafe condition before a scram or fast rod insertion is initiated; this feature enhances plant reliability and permits more freedom of action in maintenance, allowing any one set of instrumentation to be checked at a time during reactor operation. The 3 channels are described below:

i) Source Range Channel -- Two 100 curie Po-Be neutron sources located diametrically opposite each other in the core provide enough neutrons (10^4 to 10^5 n/cm^2-sec average in the core; 0.18 n/cm^2-sec at the detectors) to prevent blind reactor startup in a clean condition; as these sources decay, other sources -- primarily photoneutrons from fission product gammas interacting with the naturally occurring deuterium in the coolant -- provide an equivalent neutron flux sufficient to detect subcritical multiplication with control rods inserted. Two, highly sensitive, BF_3 proportional counters, with associated pulse integrators, log microammeters and log count rate meters, make up the low ($< 10^{-7}$ full power) flux level part of the source range channel. Higher level fluxes (10^{-9} to 10^{-4} full power) are measured by 2 fission chambers due to the lesser sensitivity of fission chambers to the gamma radiation associated with increased power. All neutron detectors are mounted inside the primary

shield tank. The channel generates fast rod insertion signals when the startup rate exceeds 1 decade/min and scram signals when the startup rate exceeds 10 decades/min. Power to the detectors in this channel is secured to minimize dissociation of the BF_3 gas and limit current to electronic circuits when neutron flux level can be adequately measured by another channel.

ii) Intermediate Range Channel -- Three compensated (for gamma flux) ion chambers with associated log microammeters measure flux levels from 10^{-6} to 1.5 times full power. This channel also generates fast rod insertion and scram signals; coincidence of two out of the three sets of instrumentation is required.

iii) Power Range Channel -- Three uncompensated ion chambers with associated linear magnetic amplifiers measure flux levels from 10^{-4} to above 1.5 times full power. Scram signals are generated at flux levels in excess of 130% full power.

g. REACTOR CONTROL SYSTEM --

Two modes of reactor control are provided: manual, for startup and low power operations; and automatic, intended for high power operation. Since manual control was found to be satisfactory for all tested conditions of plant operation, this mode is the usual one used for normal plant operation. In the automatic mode, the reactor control system positions 8 servo-controlled control rods in bank operation so as to 1) maintain constant average primary coolant temper-

ature (508F) during steady state operation, and 2) maintain primary coolant pressure within a specified range during plant power transients. The rods are driven at a rate of 3.0 to 13.5 in./min,the driving rate being proportional to the difference between reactor power level and steam demand; maximum reactivity insertion rate is limited by this driving rate to 1.5%/min, corresponding to a maximum positive startup rate of 0.8 decades/min and a weighted Doppler coefficient of $-2.3 \times 10^{-5} \Delta k_{eff}/k_{eff}$-F. Unlike a highly enriched fuel system, net control rod movement is required in transients for SAVANNAH's low-enriched, uranium oxide fuel system because of the prompt negative fuel temperature coefficient associated with the Doppler broadening of the U-238 absorption resonances; this coefficient has a greater effect for such a core than the delayed negative moderator temperature coefficient.

h. CONTROL ROD DRIVE SYSTEM --

Each of SAVANNAH's 21 control rods is driven by a twin lead screw mechanical drive actuated by a 90 volt dc,300 watt electric motor; a 3000 psig hydraulic system using high grade turbine oil reduces motor load and mechanical wear by maintaining a downward force on the control rod drive shaft to counterbalance the upward force due to primary coolant pressure; the hydraulic system also inserts all rods at high velocity (from fully withdrawn to two-thirds inserted in 0.8 sec) upon receipt of a scram signal. Three separate hydraulic supply units are provided, and each rod has its own

independent hydraulic accumulator, to ensure availability of hydraulic energy for scram protection. This design ensures availability of scram protection at any ship heel angle, including capsized. Rod drive motors are run normally by rectified 450 volt ac power, but in emergencies by a nickel-cadmium battery with sufficient power to drive all rods at least 5 minutes, long enough for normal insertion. In such an emergency, a lead-acid battery provides power to the rod control system. Buffer seals are provided where the rod drive shafts penetrate the reactor vessel head; by ensuring a continuous, inward flow of purified coolant, these seals prevent primary coolant leakage from these penetrations. Failure to be able to correct persistent hydraulic oil seepage in the hydraulic system resulted in use of an inert (nitrogen) atmosphere in the containment vessel to eliminate the hazards of potential oil fires or explosions. Rod-position indication is provided by a magnet affixed to the rotor of each drive motor; each turn of the rotor -- corresponding to 3/8 in. rod travel -- closes a reed switch which feeds an electromechanical counter at the control console. "Rod bottomed" indication is provided by a magnet mounted on the top of the control rod drive lead screw, which closes a reed switch mounted outside the pressure tube in which the lead screw travels; the reed switch energizes a "rod bottomed" light on the control console. A gyro-activated capsize switch is provided to irrevocably drive in all rods fully if the ship's heel angle exceeds 45 degrees for more than 5 seconds.

i. <u>REACTOR SAFETY SYSTEM</u> --

In addition to the above-described scrams and fast rod insertions initiated by the nuclear instrumentation system, this provides reactor scrams for the following conditions:

a) primary coolant temperature greater than 540F (this provides a backup for the high power scram)

b) primary system pressure less than 1200 psi (this prevents steam blanketing and potential resultant burnout of fuel pins)

c) loss of power to all 4 primary coolant pumps (this prevents core damage if primary coolant flow is lost)

d) low hydraulic supply manifold pressure (this prevents reactor operation without completely reliable scram protection available)

e) operator action

4. <u>POWER PLANT DESCRIPTION; PROPULSION SYSTEM</u> --

The propulsion machinery on SAVANNAH is essentially the same as that on a conventional steam-powered ship: a two-element steam turbine driving a single propeller through mechanical reduction gears. The propulsion system contributes 1,265 short tons to the total 4,348 short tons power plant weight; the remainder is made up of reactor system (1,665 short tons) and shielding (2,418 short tons). The only unique feature of the engine room resulting from use of a reactor plant for steam generation is the consolidated control room.

Emergency propulsion power is supplied by a 750 hp electric motor which engages one of the high-speed pinions in the reduction gear via a quick-connect coupling. An oil fired boiler is provided for generation of air ejector motive steam to maintain a main condenser vacuum during emergency propulsion operation, thereby reducing blade windage losses in the main turbine. Figure A-10 is a heat balance for the plant.

Normal electric power is supplied by two, geared, steam-turbine generator units; standby electric power is furnished by two diesel generators, and emergency power by another diesel generator located on the navigation bridge deck. The secondary system is shown schematically in Figure A-7. Two 16,000 gallon per day distillers of the multiple-effect type provide ample fresh water for plant makeup (via ion exchangers), drinking, washing and culinary needs. The various portions of the propulsion system are described in detail below:

a. THE MAIN PROPULSION UNIT --

A cross-compound turbine with high- and low-pressure sections directly coupled to a double-helical, double-reduction gear of conventional design, this unit delivers 22,000 maximum SHP at 110 rpm with saturated steam at a pressure of 472 psia and a condenser vacuum of 28.45 in. Hg; astern power is 8,000 SHP at 53.5 rpm, in compliance with standard practice of 80% of normal ahead torque at half rpm with 100% normal ahead steam flow. The propeller is five-bladed, made of nickel-manganese-bronze. The 4,500 rpm,

high-pressure turbine has 9, single-row stages of impulse type blading. The 3,000 rpm, low-pressure turbine has 7, single-row ahead stages of impulse type blading; this turbine also has 1 double-row and 1 single row stage of impulse type blading for astern operation. Both turbine casings are split on a horizontal plane. High-pressure turbine exhaust steam, with 11% moisture content, passes through a 2-stage, baffle/cyclone, moisture separator before admission to the low-pressure turbine; inter-stage moisture collecting provisions are also included in both turbines. Steam flow is regulated by an electric motor-operated throttle valve controlled by an electrical servo system mounted on a maneuvering handwheel on the main control console.

b. CONDENSERS AND FEEDWATER SYSTEM --

Hung from the low-pressure turbine, the main condenser is a single pass, non-divided design with scoop sea water injection for normal operation and a 150 hp, 20,000 gpm sea water circulating pump for standby and maneuvering. The 3/4 in. copper-nickel tubes are welded to single, copper-nickel tube sheets for tightness and then lightly rolled for vibration resistance. The 2 turbine-generator condensers are two-pass design, also with welded tubes, and are cooled to maintain 29 in. Hg vacuum by continuous duty sea water circulating pumps. Feedwater is drawn from condenser hotwells by two 40 hp condensate pumps, heated and deaerated and returned to the steam generators at 347F by the 650 gpm (maximum) steam-turbine driven feed pump.

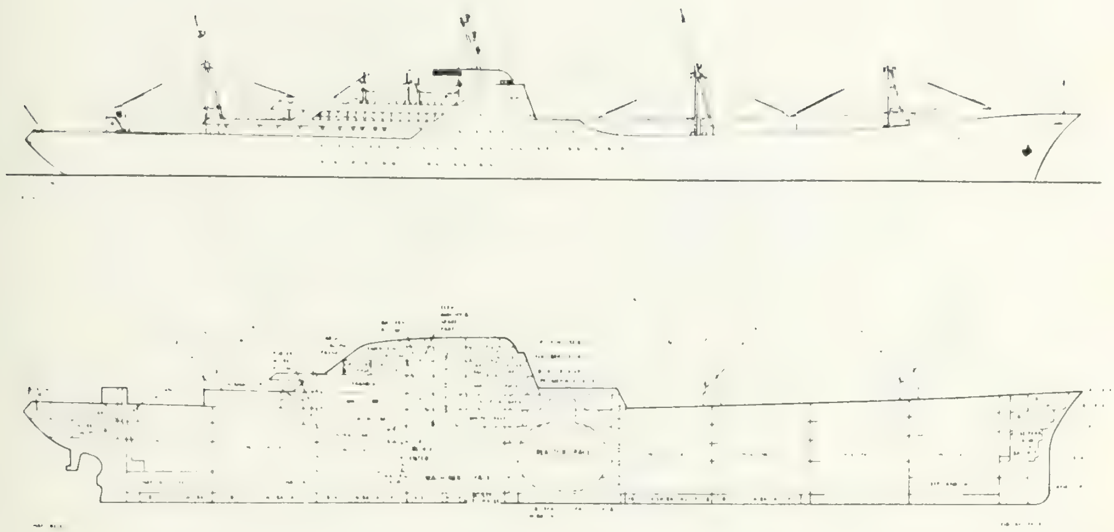
c. ELECTRICAL SYSTEM --

SAVANNAH's 5 electrical generating units are all 450 volts, 60 Hz, 3 phase; two 1,500 kw main turbine-generators, two 750 kw auxiliary diesel generators and one 300 kw emergency diesel generator are provided. The 1,500 kw units are 8 stage, impulse turbines driving 1,200 rpm generators through compact planetary gears. The two 750 kw units provide power for: 1) decay heat removal following reactor shutdown, 2) emergency, "take home" propulsion power if the nuclear power plant should fail, 3) reactor startup, and 4) spare generating capacity. Both auxiliary diesel units start and come on the line automatically upon loss of normal electrical power. The emergency diesel unit provides power to a 450 volt emergency switchboard and will operate only when all other 4 generators are secured; the emergency switchboard supplies such vital loads as primary coolant pumps (half speed only), emergency cooling, and emergency lighting. The electrical distribution system is shown in Figure A-8. High reliability is obtained by use of a split-bus arrangement with the two busses connected by a normally closed bus tie-breaker; in the event of a bus fault, the tie-breaker opens and automatic bus transfer switches transfer essential loads to the unfaulted bus.

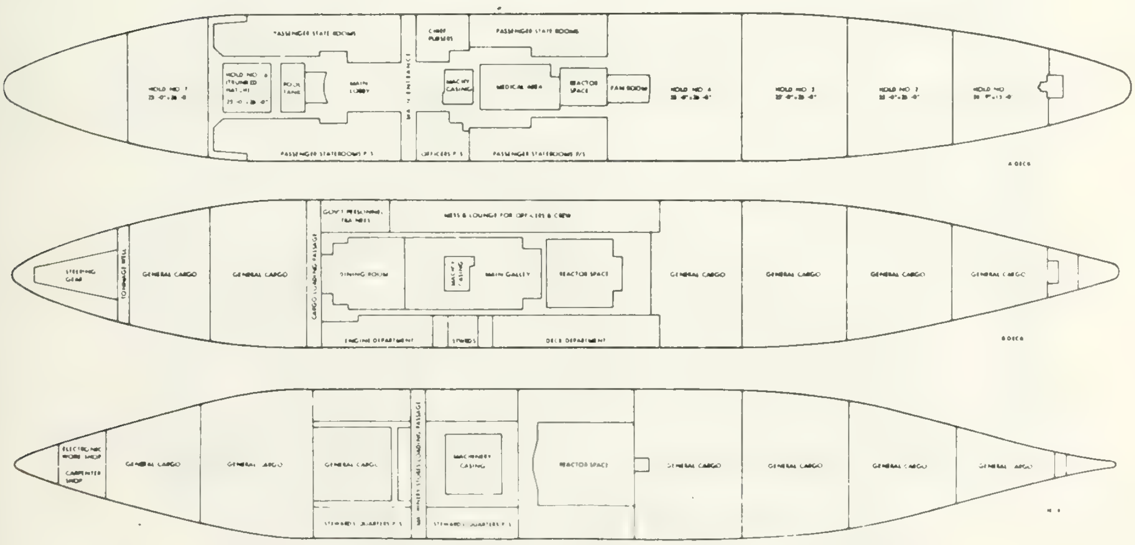
5. NUCLEAR SERVICING VESSEL, NSV ATOMIC SERVANT --

A description of N.S. SAVANNAH would not be entirely complete without some mention of her specially built servicing vessel, a non-propelled craft 129 ft long with a 36 ft beam.

Shown in Figure A-9, the vessel is equipped to maintain, service and refuel SAVANNAH's reactor system, including facilities to handle, process and package all radioactive waste from refueling, by providing those required facilities not normally found in a conventional shipyard or drydocking facility. It can store 18,400 gallons of liquid with activity levels up to 1 microcurie/liter and can reduce higher activity levels in liquids by filtration, ion exchange or dilution. It can store the entire 53 cu ft of spent resin from SAVANNAH's 3 purification system demineralizers with average activity levels up to 6 curies/liter, and can package these wastes in concrete drums for disposal. ATOMIC SERVANT can store all 32 spent fuel elements and all 21 control rods and can handle, store and decontaminate other items as large as primary coolant pumps. Full length longitudinal bulkheads, 8 ft in from the sides, are installed to protect the fuel storage pit in the event of a collision.

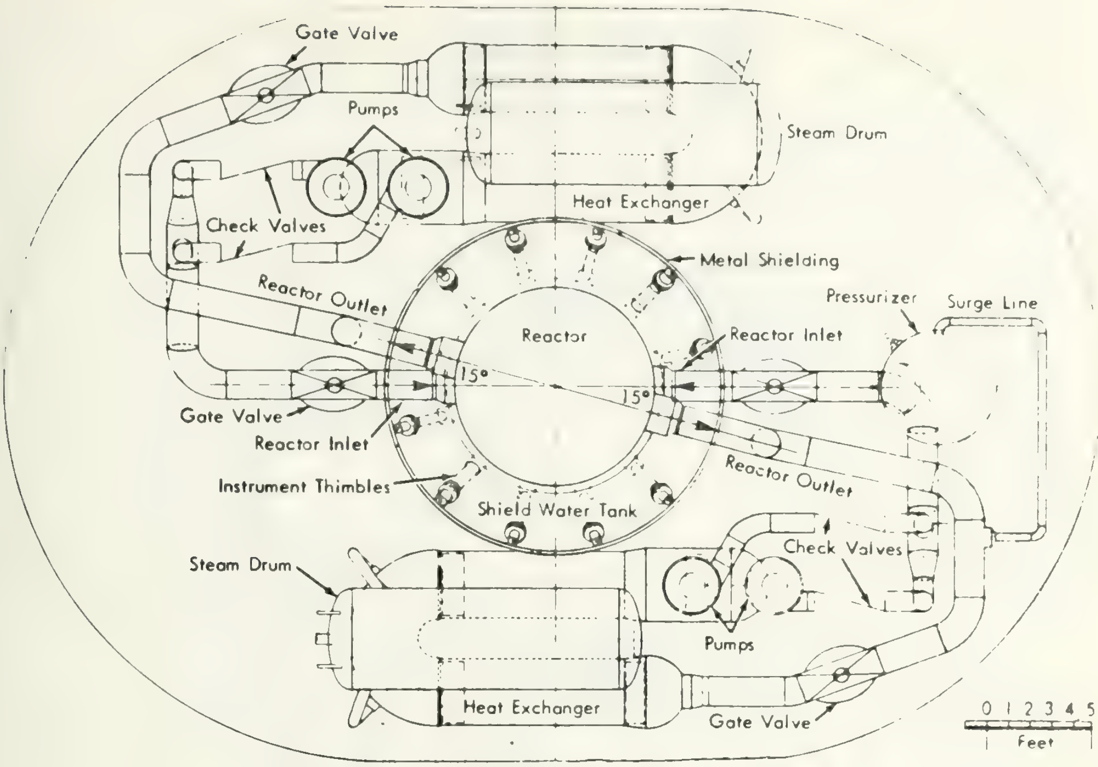


Outboard and inboard profiles of the N.S. *Savannah*.

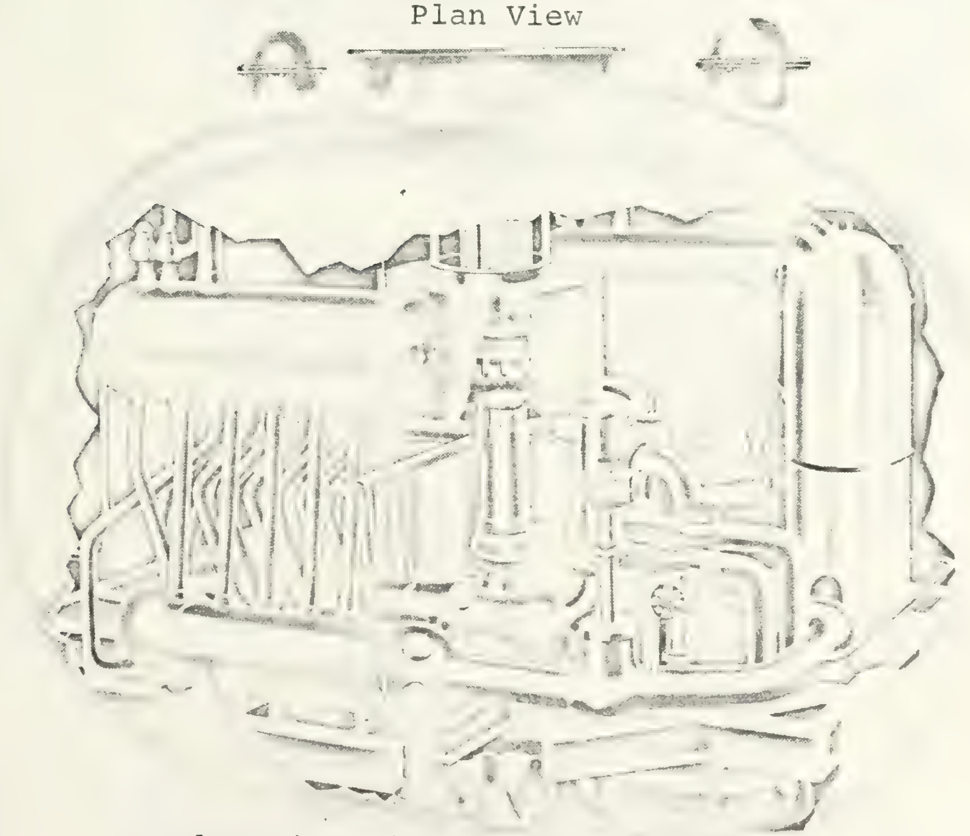


Deck plans

Figure A-1. N.S. SAVANNAH General Arrangement



Plan View



Elevation View

Figure A-2. Containment Vessel Arrangement
N.S. SAVANNAH

00202

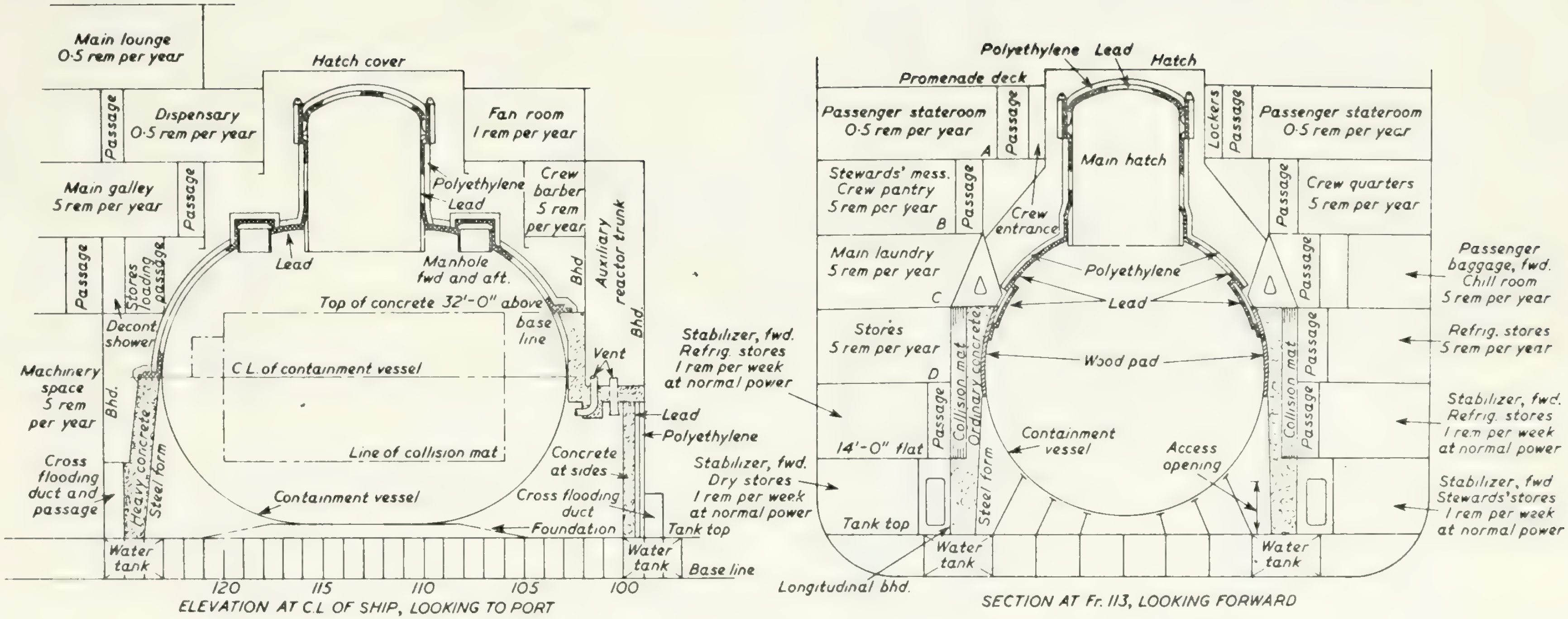


Figure A-3. Secondary Shielding and Design Radiation Dose Rates
N.S. SAVANNAH

00203

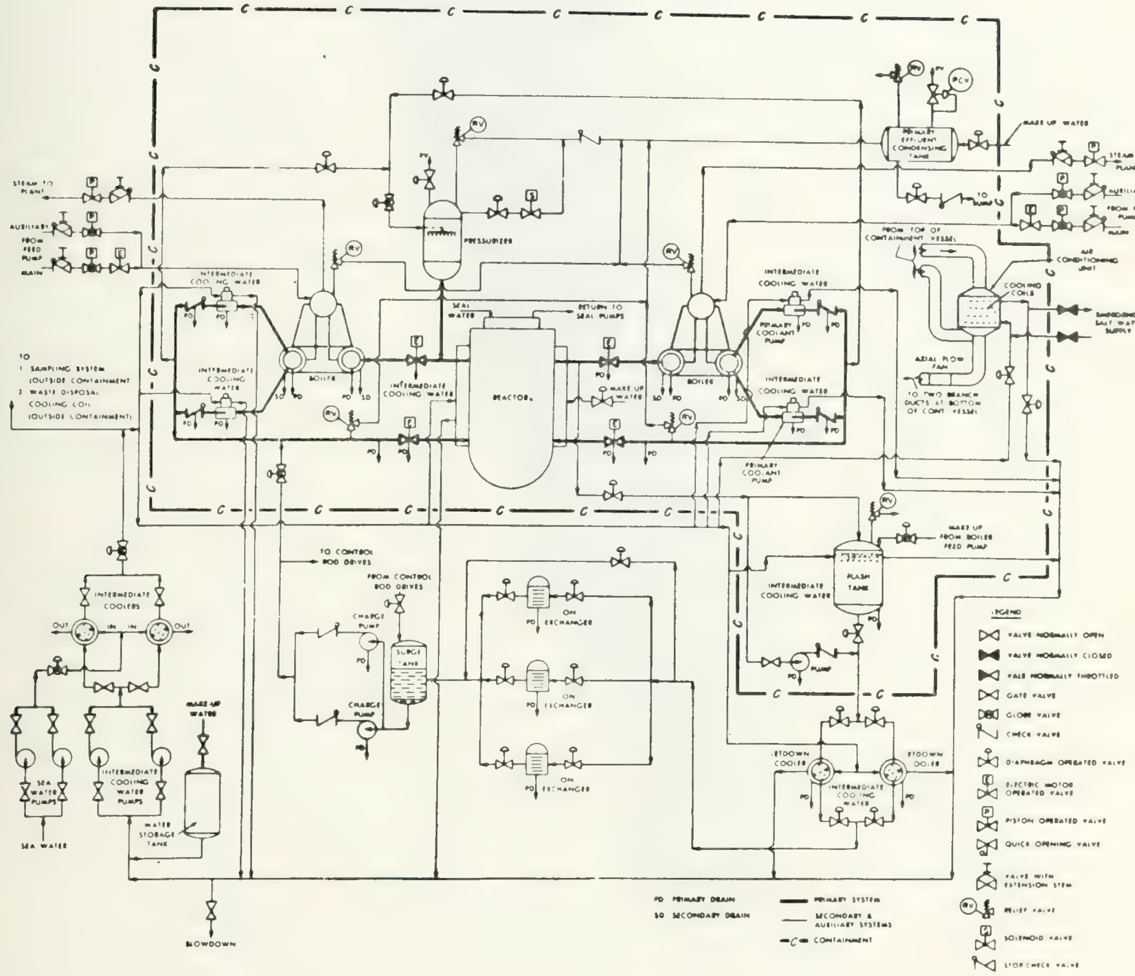


Figure A-4. Primary System Flow Diagram

N.S. SAVANNAH

00204

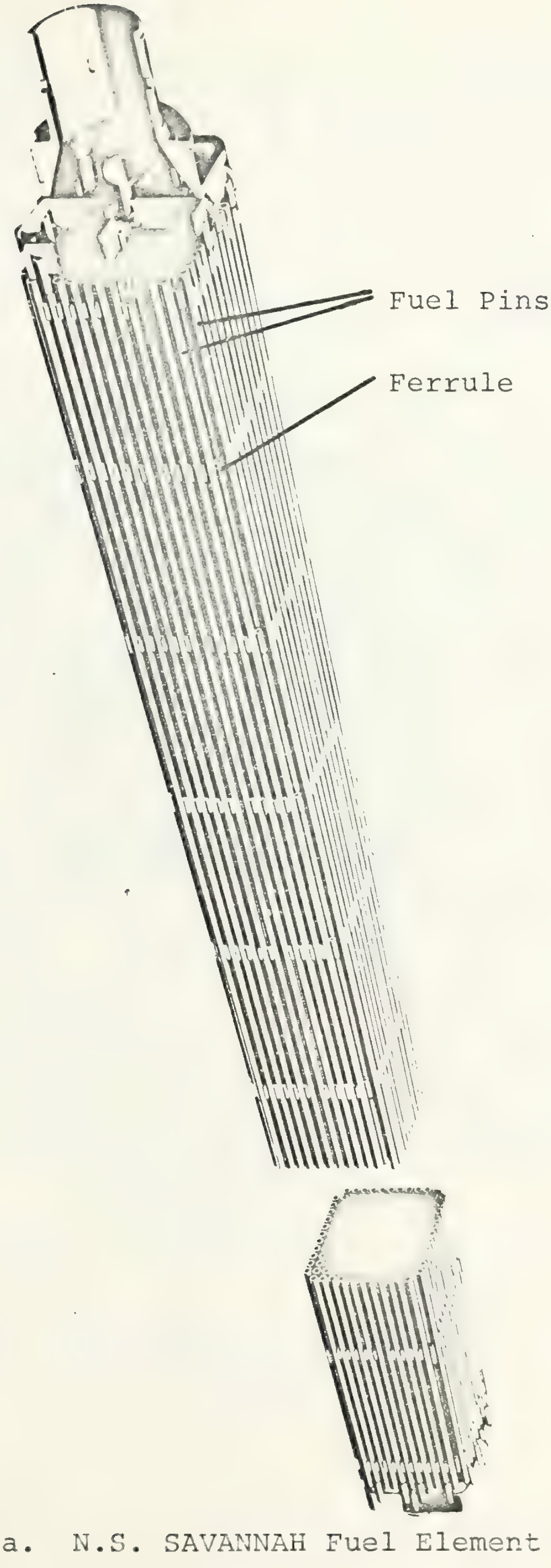


Figure A-5a. N.S. SAVANNAH Fuel Element

00205

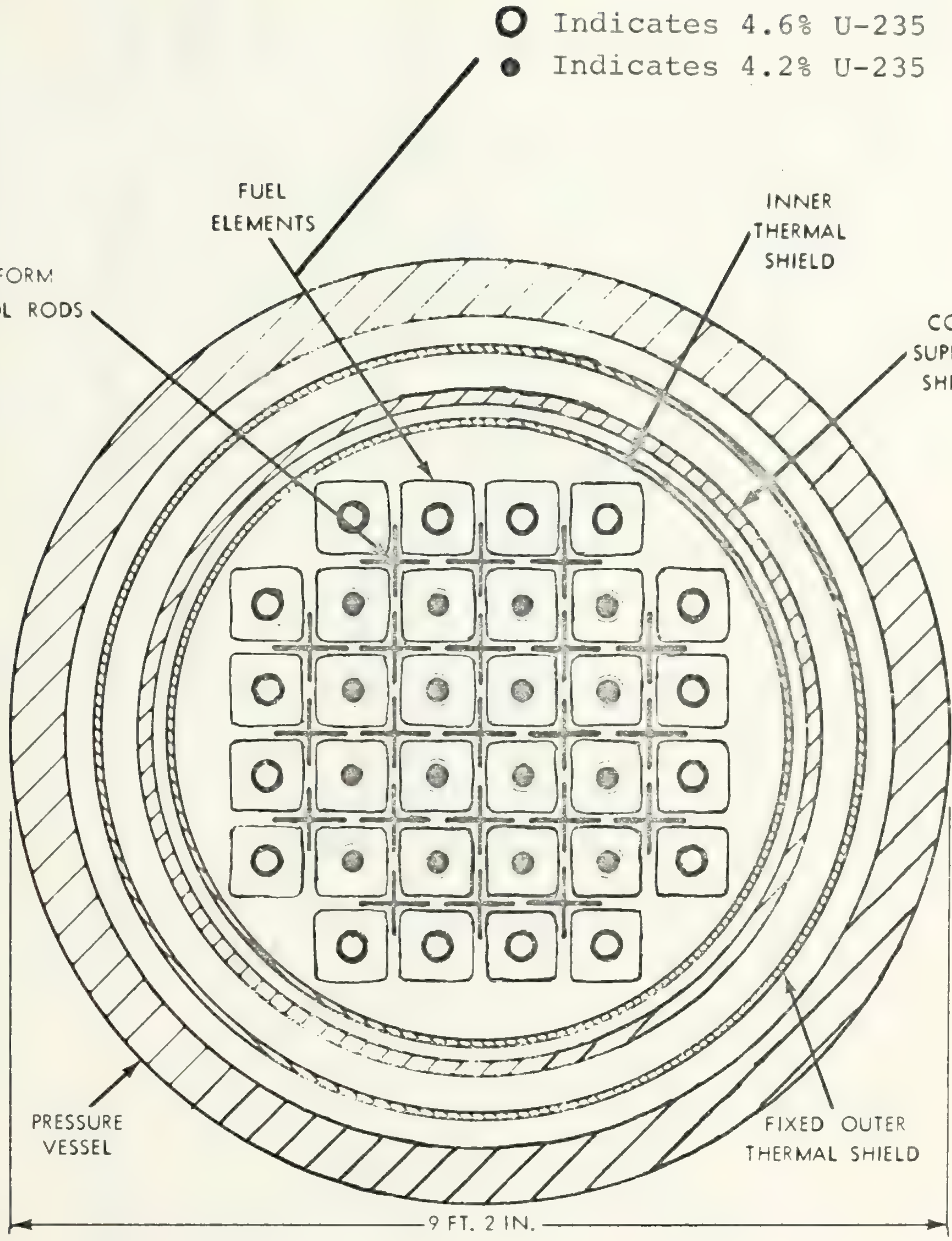


Figure A-5b. Reactor Core Cross Section
N.S. SAVANNAH

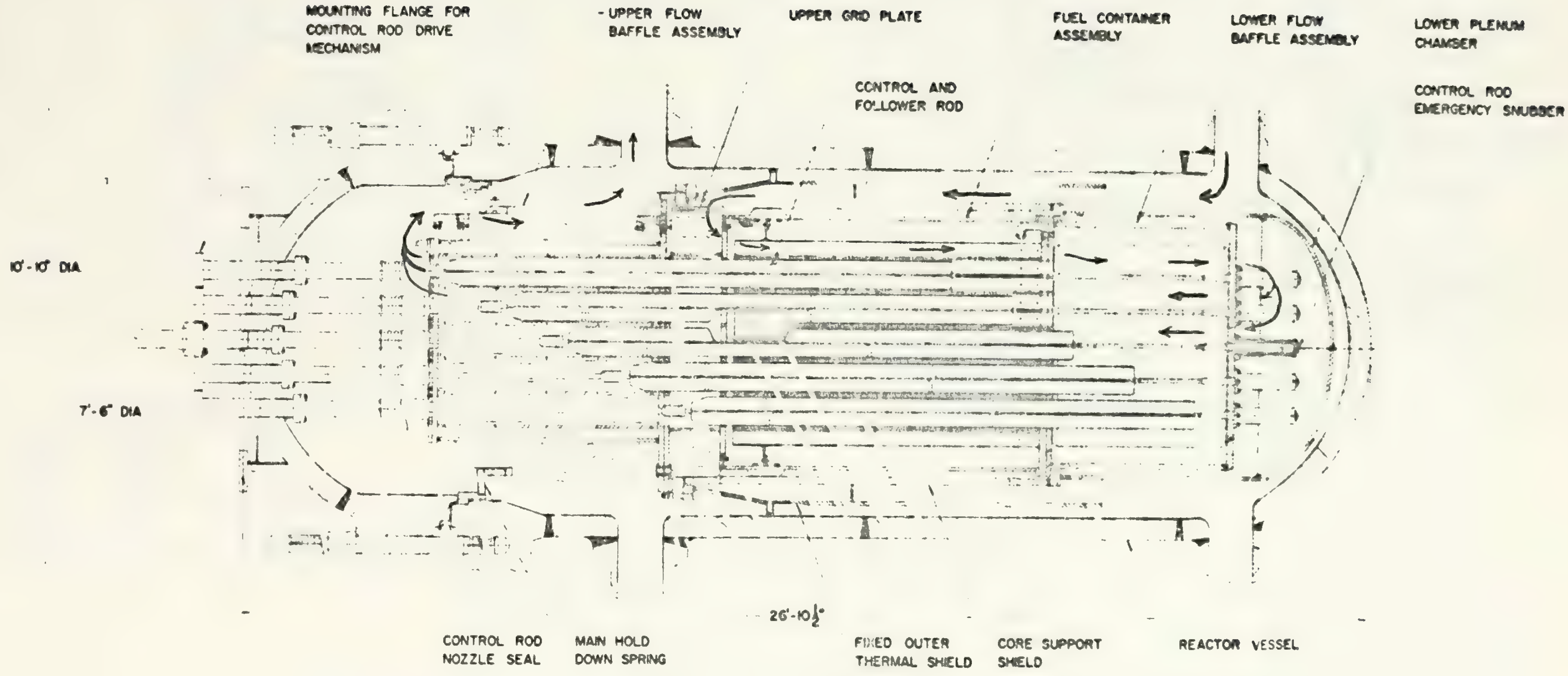


Figure A-6. Reactor Core Vertical Section
N.S. SAVANNAH

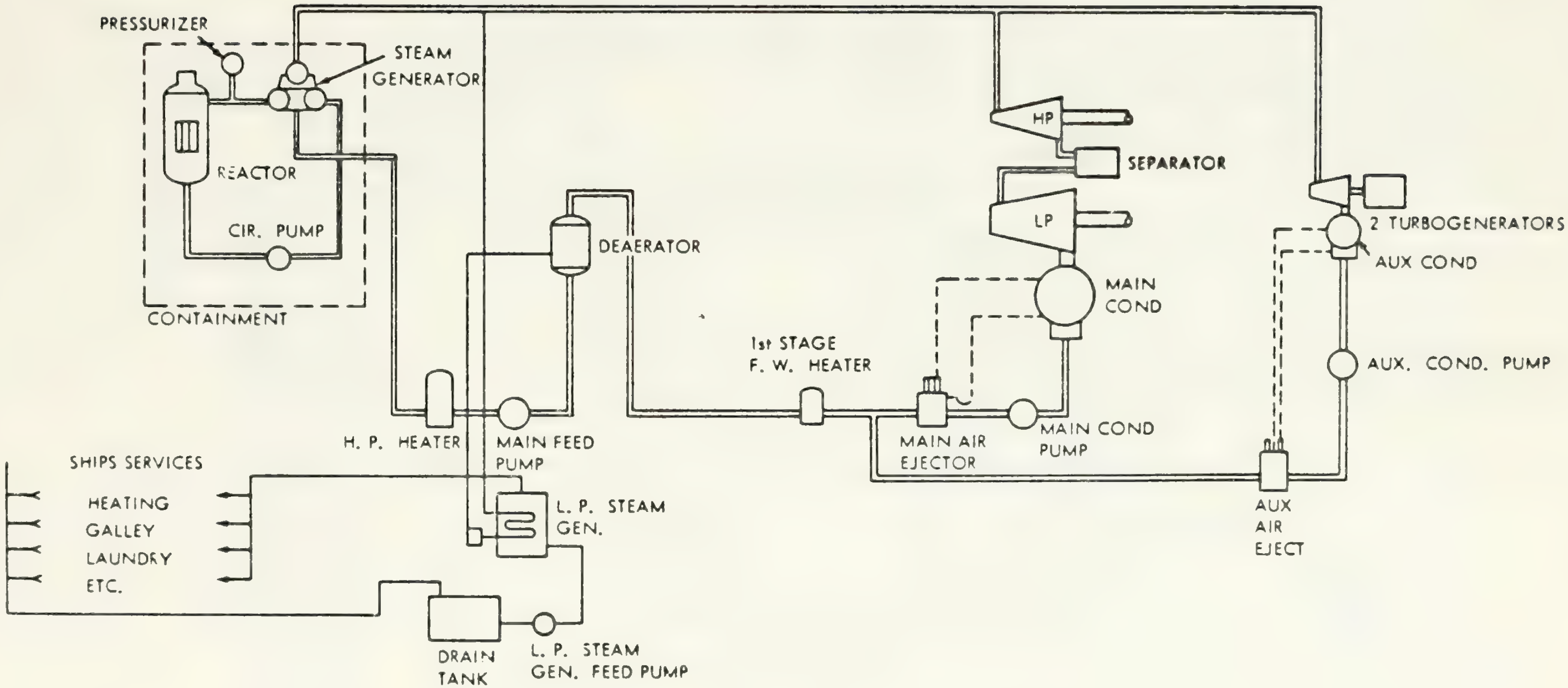


Figure A-7. N.S. SAVANNAH Secondary System Flow Diagram

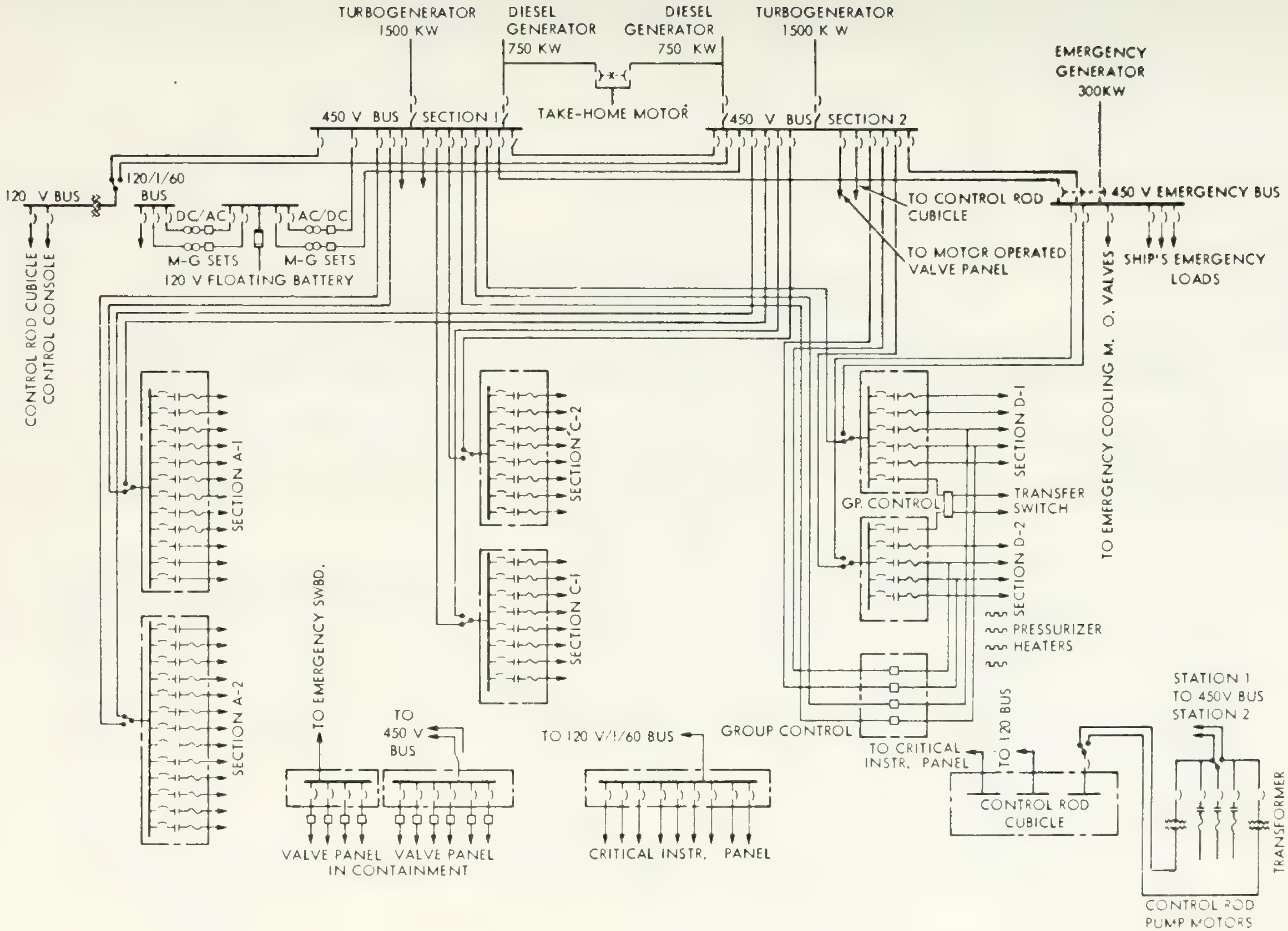


Figure A-8. N.S. SAVANNAH Electrical Distribution System Diagram

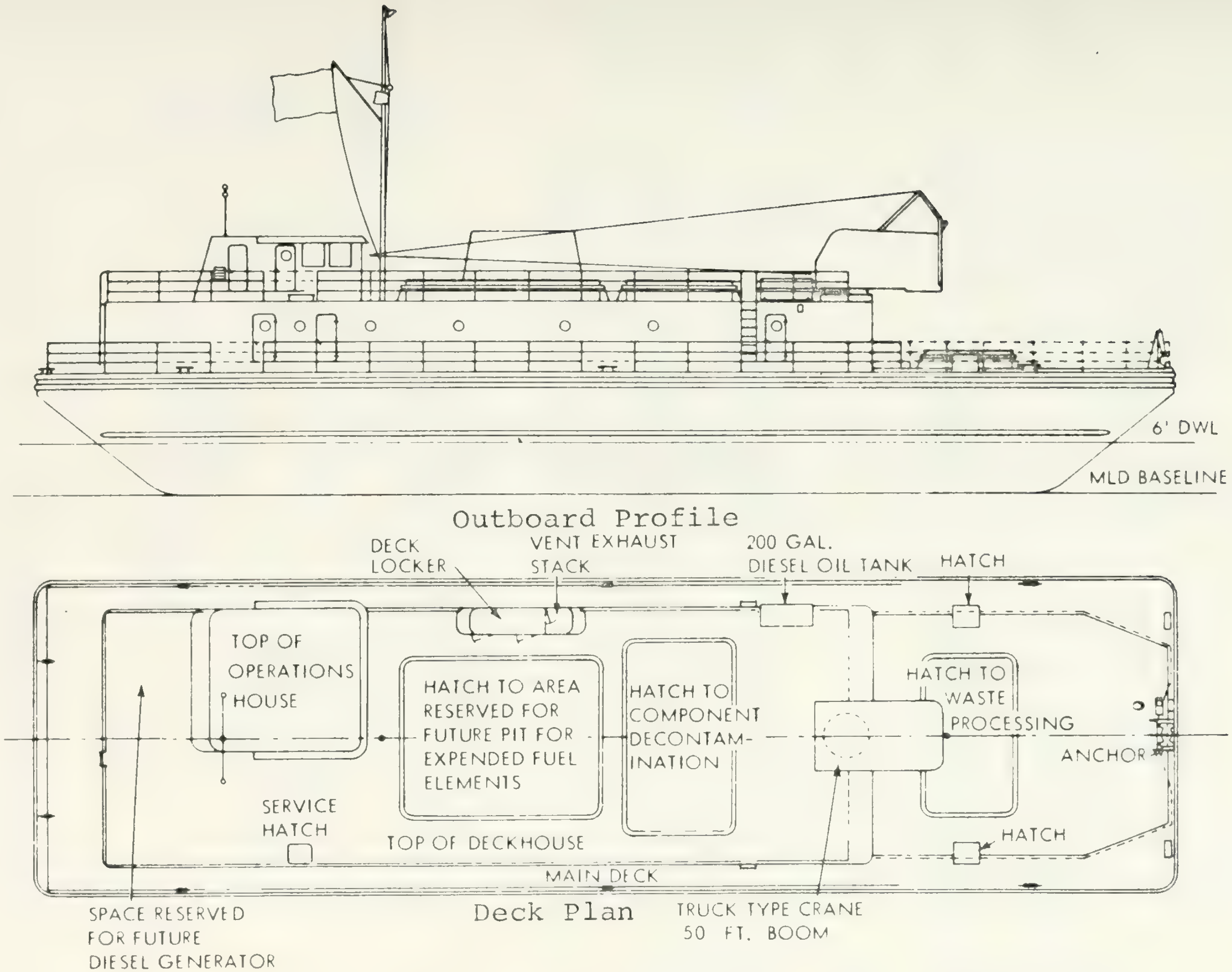


Figure A-9a. N.S.V. ATOMIC SERVANT General Arrangement

00210

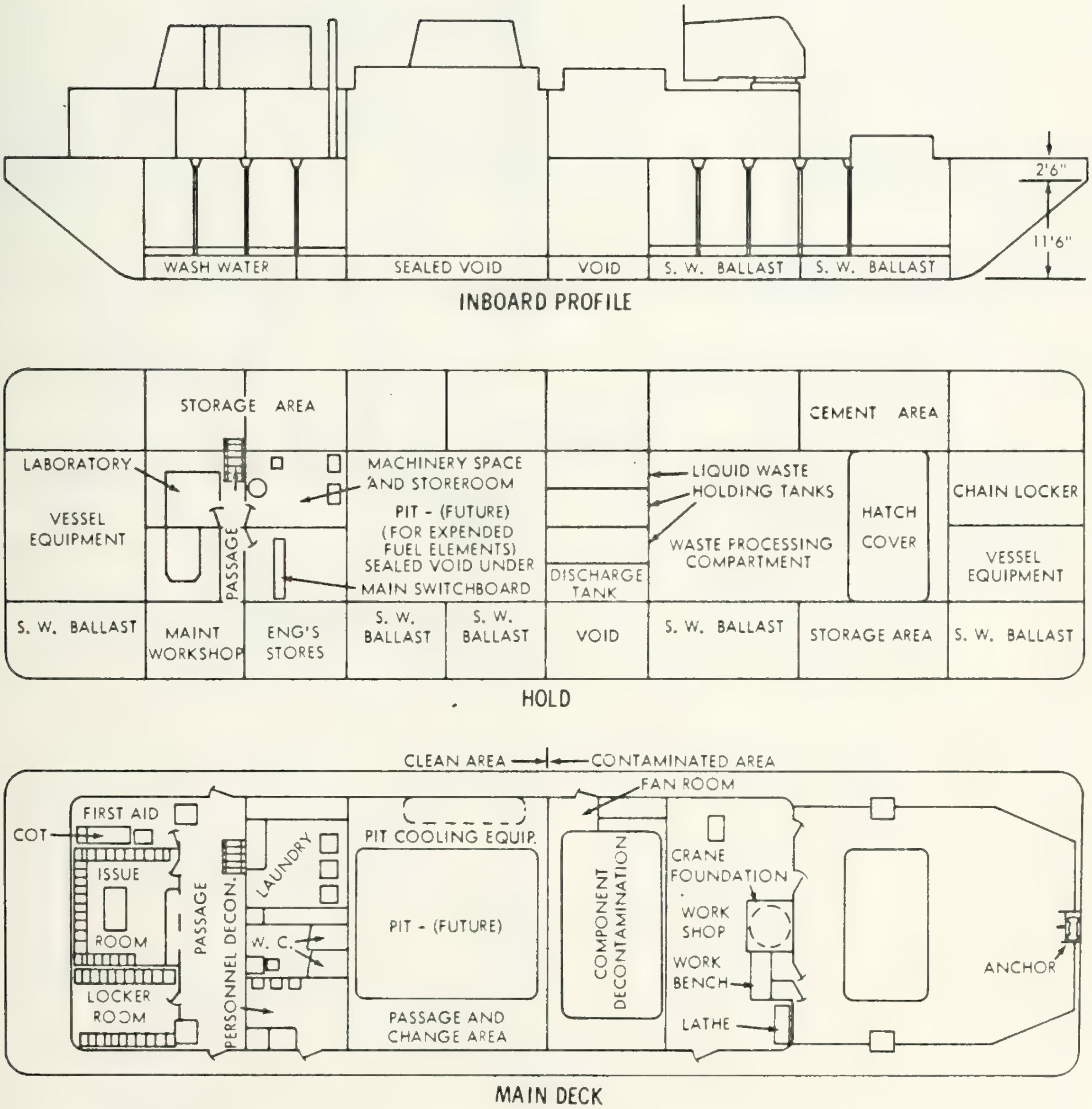
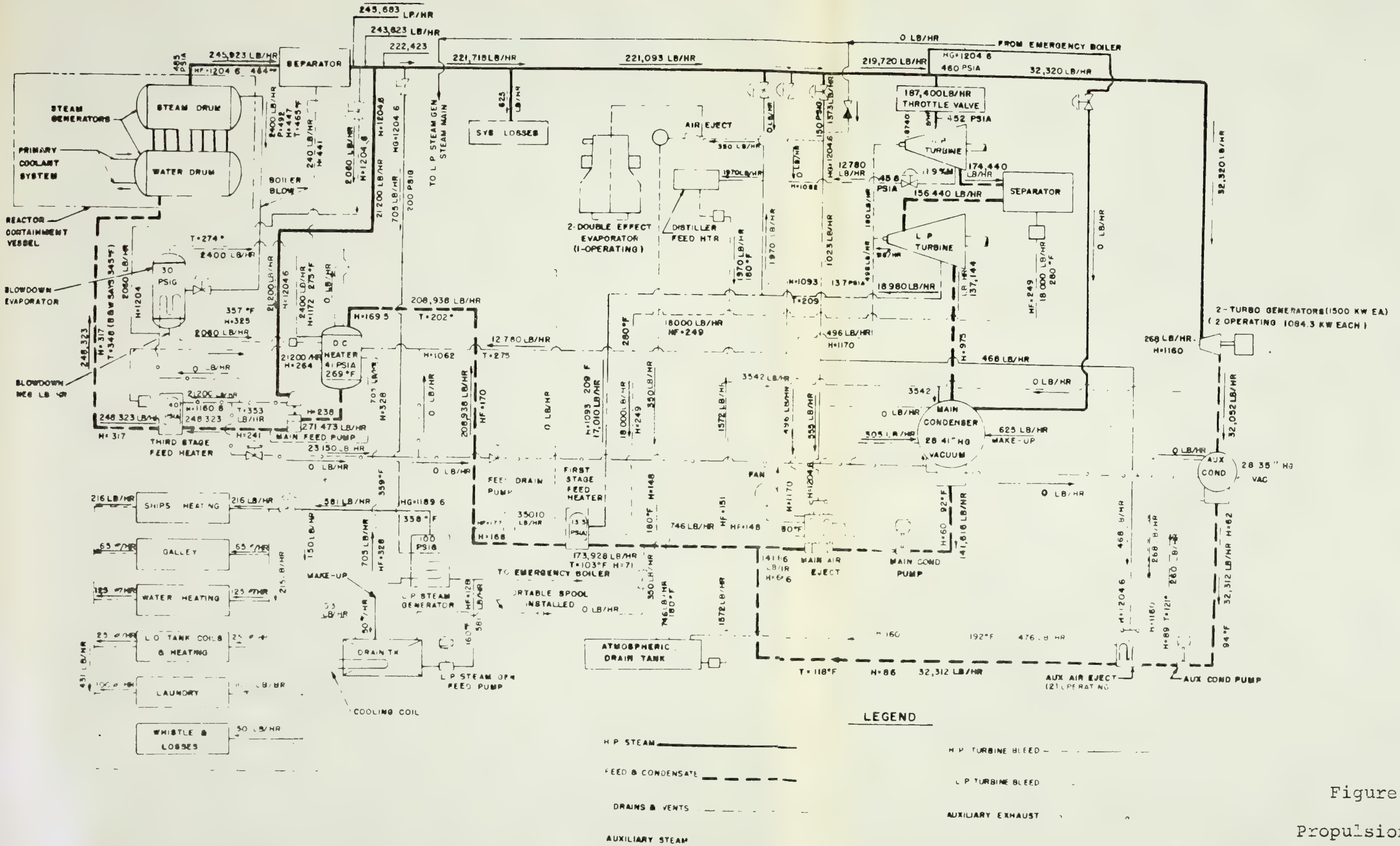


Figure A-9b. N.S.V. ATOMIC SERVANT General Arrangement

00211



SHAFT HORSEPOWER (NORMAL)	20,000
SHAFT HORSEPOWER CALCULATED	20,040
HEAT OUTPUT BOILERS, MEGAWATTS	64.03
PROPELLER RPM	107
TURBINE WATER RATE EXTRACTING, LB/SHP-HR	9.35
EVAPORATOR LOAD GPD	9,760
GENERATOR LOAD KW INCLUDES STABILIZER LOAD	2168.7
SEA WATER TEMPERATURE °F	75°F
QUARTERS AIR CONDITIONING	IN USE
QUARTERS HEATING SYSTEM	IN USE
MN TURBINE WATER RATE NON EXTRACTING	8.59 LB/SHP
MN TURBINE THROTTLE PRESS. & TEMP	SAT 460 PSIA
MAIN COND'R VAC	28.45 HG

BLEEDER PRESSURES GIVEN ARE THE ANTICIPATED PRESSURES AT TURBINE FLANGES

Figure A-10 N.S. SAVANNAH

Propulsion Plant Heat Balance

B. N.S. OTTO HAHN (ref's. 28 through 39, 71)

1. GENERAL --

OTTO HAHN is a single-screw, 13 compartment, nuclear research ship/ore carrier/passenger ship of 14,000 tons deadweight and 25,812 tons full load displacement at a draft of 30 ft 2 in. Design speed is 15 3/4 knots at 10,000 SHP and 38 MWt reactor power. Other principal characteristics of OTTO HAHN are as follows:

Length, Over All/Between Perpendiculars	564 ft 3 in./515 ft 1 in.
Beam, molded	76 ft 8 in.
Freeboard	17 ft 6 in.
Block Coefficient	0.741
Cargo Deadweight/Capacity	14,000 tons/468,000 cu ft
Ballast Water Weight/Capacity	14,700 tons/590,000 cu ft
Crew	66
Passengers	47
Standard of Subdivision	3 compartment

Since ore carriers are weight-limited ships, violent rolling in waves would result if the dense cargo were placed low in the ship; this undesirable and dangerous condition is precluded by loading the ore on top of deep tankage, used to hold water ballast for stability in the unloaded condition, high enough to achieve a suitably small metacentric height ($\overline{GM}$). OTTO HAHN's $\overline{GM}$ in the ballasted and full load conditions is between 2 ft 5 in. and 2 ft 11 in., resulting in a roll period of between 10 and 20 seconds. This required tankage provides

a ready means for ballasting the ship to the design full-load draft to facilitate performance of full power trials in the unloaded condition.

A high navigation bridge forward facilitates ship handling in restricted waters, thereby reducing the danger of collision and grounding. The unusually large amount of living space aboard the ship will accommodate crews in training for operation of this and other nuclear ships. All recommendations of the 1960 Safety of Life at Sea Convention have been complied with in the accommodations for crew and passengers. The 3 compartment standard of subdivision used is 1 compartment in excess of that required by the rules; this greater degree of subdivision results in increased ship safety.

2. <u>SHIP ARRANGEMENT AND STRUCTURE</u> --

The general arrangement of the ship is illustrated in Figure B-1; the hull is subdivided by 13 watertight bulkheads. The 4 forward ore holds are separated into two groups by an auxiliary engine room which houses an auxiliary diesel generator set, an auxiliary switchboard, and ship fire fighting systems. Between these forward holds and the after two holds is the nuclear propulsion plant; as illustrated in Figure B-2, the propulsion plant is subdivided into 7 watertight compartments, 2 of which (cofferdams) separate the 3 nuclear spaces from the rest of the ship. In addition to enhancing safety, these 2 cofferdams permit cutting the ship into 3 floatable parts for later replacement of the reactor portion with a more advanced design. Control panels for the

reactor and propulsion machinery are located in a central control room above the main engine room.

Only the main engine room and the auxiliary boiler room extend the full width of the ship; all other compartments are separated from the ship outer hull by longitudinal side tanks which can be filled with water ballast and which are cross-connected by flooding pipes to assure transverse symmetric flooding for improved stability in the damaged condition. In conformance with the rules of German Lloyd and Bureau Veritas for nuclear ships, the tanks beside and beneath the reactor area are kept free of load (including ballast water) and machinery; the side tanks each span 0.244 times the ship's beam from the outer plating at each side of the reactor spaces (the rules require at least 0.2 times the beam). One way valves in this tankage allow flooding water to flow from side tanks to bottom tanks for increased ship stability in the damaged condition. Rather than provide a land-based support facility, a reactor service room has been built into the ship to provide nearly all facilities and services necessary to support any reactor servicing and refueling operation.

The double bottom in the vicinity of the reactor spaces has been specially designed to withstand grounding with minimal reactor damage. Normal double bottom depth of 5 ft has been increased to 8 ft 3 in. and an intermediate, horizontal, watertight plate has been inserted, with vertical stiffeners above and below this plate offset and designed so

that plate buckling will reduce transfer of grounding shock to the containment vessel and its contents. Collision protection is enhanced by extensive stiffening of the ship sides along the reactor area. As shown in Figure B-3a, decks, frames and other stiffening members have been added to the customary ship structure to absorb collision energy; the entire collision barrier, including hull plating, main deck, and inner bottom plating in this area, is made of a special, tough grade of steel, B.V. grade "ESS". Detailed calculations and 1:7.5 scale destructive testing indicate that only a very few ships at full speed would be able to penetrate the reinforced barrier to impair the integrity of the longitudinal bulkheads bounding the reactor area.

3. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

a. Containment Vessel --

The 167 ton containment vessel, shown in Figures B-3a and B-3b, is a 29 ft ID, vertical cylindrical shell with hemispherical top and ellipsoidal bottom. Shell plating and bottom are BH51 (Ruhrstahl AG Heinrichshütte Hattingen) steel with thicknesses of 1.18 in. and between 1.18 and 1.57 in. respectively; the top is BH36 steel with a thickness of 0.78 in. Entry to the 42 ft 7 in. high vessel is via a lock in the top; a bolted cover of 17 ft 8 in. diameter is provided for refueling access. All access to and from the containment vessel and the reactor auxiliary room is by way of a laboratory equipped for radiation monitoring and decontamination. The cylindrical shell portion contains the 25 piping

and the 74 electrical cabling penetrations, plus 4 flooding valves that open at a depth of 98 ft to prevent vessel collapse in the event of sinkage in deep water. Cable penetrations consist of unsheathed cables in fused glass sheets set in frames welded into the vessel wall. The vessel contains the radioactive systems and fluids except for long-term, low activity level storage tanks located in the reactor auxiliary room; it is designed to prevent release of radioactivity in the event of a major primary system boundary rupture (258 psia internal vessel pressure at 392F). Measured gas leakage rate from the containment vessel at 9 atmospheres internal pressure indicates the leakage rate at 18 atmospheres (265 psia) is 0.17 volume percent/day; the design criterion at this pressure is less than 1.0 volume percent/day.

Vessel support is provided by 24 brackets welded to the bottom; these brackets rest on mating brackets welded to a conical ring which transmits the weight of the vessel and its contents (approximately 1000 tons) to the top plating of the ship's upper double bottom. These brackets are set at an angle of 45° to the horizontal and are faced with low-friction Teflon plates and connected with necked-down bolts to act as a large ball-and-socket joint and to allow for thermal expansion. At the height of its center of gravity the vessel is connected at 4 points to the bulkhead corners by means of 4 sets of 16 each long, prestressed tie rods for additional lateral and longitudinal support. The 150 ton (for each group of 8 tie rods) prestress is not completely

removed under any ship attitude, thereby maintaining vessel position laterally and longitudinally.

b. Radiation Shielding --

The primary shield, shown in Figure B-3b, consists of: a closed, neutron shield tank 17 ft 4 in. OD by 16 ft 4 in. high, the center of which is occupied by the pressure vessel; 10 to 12 in. thick annular layers of cast iron around the pressure vessel; and top and bottom cast iron cover plates. Where the neutron shield tank is recessed to accommodate the primary coolant pumps, a compensating cast iron shield is inserted in the tank. Since the design dose rate outside the primary shield is 20 mrem/hr, the containment vessel is accessible, if necessary, for short periods during reactor operation.

Periodic inspection of the containment vessel exterior surface as required by ship classification society rules is permitted by mounting the secondary shield on the surrounding inner hull and the stiffened bulkheads of the reactor room; to prevent stiffening interaction between the shield and the ship's steelwork, which could lead to unacceptable stresses in the steel, the shield is installed over 1 1/4 in. thick slabs of foil-wrapped, expanded polystyrene. This shield, also shown in Figure B-3b, consists of a 20 to 24 in. thick square wall of 203 lb/cu ft density concrete, on top of which is mounted a cast-concrete truncated cone closed by a removable concrete cover. The corners between the walls and the cover are covered with steel plates through

which pass the piping and cabling connecting the containment vessel to the engine room.

The secondary shield is designed to reduce radiation levels in adjacent spaces enough to permit normal occupancy/working time in these spaces: i.e., below 0.02 mrem/hr in living quarters, and below 0.06 mrem/hr in the control room and machinery spaces. These very low levels are achieved by keeping radiation levels below 0.2 mrem/hr at the surface of the secondary shield. An additional design criterion used for the secondary shield is that the integrated one week radiation dose in the control room must be below 25 rem if the core melts down in such a way as to release and uniformly distribute in the containment vessel 100% of the fission gases and 0.3% of the solid fission products. It weighs 1,100 tons.

c. Primary System --

The primary system is shown in Figure B-4. Installation of the steam generators inside the pressure vessel and the use of short, concentric piping to the 3 primary coolant pumps resulted in significant reduction of the extent of primary coolant boundary required. Elimination of the need for a separate pressurizing system further reduced the primary boundary; pressure control is effected by operation at a primary pressure of only 925 psig, thereby maintaining a free water level in the pressure vessel with a saturated steam space at the top of the vessel. An advantageous result of this consolidation of components is a significant decrease both in the amount of secondary shielding required and in the

potential for primary system rupture; the latter greatly reduces the probability of a loss of coolant accident. The slight boiling in the core necessary to maintain the pressure vessel steam bubble makes the OTTO HAHN reactor somewhat of a compromise between a pressurized and a boiling water reactor as far as core design is concerned.

Each of the 3, canned-motor, 15 kw, simple axial impeller, primary coolant pumps is arranged vertically alongside and connected by a single pipe nozzle to the bottom head of the pressure vessel; the pipe nozzle consists of 2 concentric pipes to accommodate both incoming and outgoing coolant flow. As shown in Figure B-4, these pumps pull the light water primary coolant down in an outer annulus in the pressure vessel over the steam generator tubes and past the thermal shield to the pump annulus; the pumps then deliver the cooled water through the inner concentric pipes to the plenum chamber beneath the single-pass core. Flowing upward through the core, the coolant is guided by a chimney to within 9 in. of the water-steam interface where it turns downward to flow again through the steam generators. Full power primary flow rate is 13,160 gpm, yielding a flow velocity through the core of 5.6 ft/sec; core inlet temperature is 532.4F and total pressure drop from pump outlet to inlet is 3.7 psid. The inner (discharge) pipe of each pump is fitted with a partially closing, hinged flap that prevents significant backflow through a secured pump. In the event one pump is inoperable, the plant can be operated up to 74% full power using the

remaining two pumps.

Unlike in the SAVANNAH design, the location of the OTTO HAHN steam generators rather high above the core provides sufficient natural circulation of coolant for reactor operation up to 11 MW, 29% full power; this, coupled with the arrangement of the primary coolant pumps and piping, makes a cold water accident an extremely remote possibility. The steam generator is sufficiently removed from the core (approximately 3 ft, edge-to-edge) that N-16 gamma activity in the steam leaving the containment vessel is extremely low. The steam generator is divided into 3 parallel sets of 54 each, single-pass tubes; each set has its own independent, external feedwater and steam isolation valves for use in case of tube leakage. Individual tubes can be plugged and seal-welded in the conventional manner, after removal of the associated flanged and bolted piping connections, from outside the vessel. The tubes are made of Inconel with an OD of 0.767 in. and a wall thickness of 0.047 in. Heat exchange area is 5,000 sq ft for all 3 sets of tubes combined. Feedwater enters the tubes at 365F and leaves as 65F superheated steam at 523.4F, 456 psig; full power flow rate is 128,000 lbs/hr. With the reactor vessel upper head off, the entire steam generator can be removed by unbolting the tube sheets from the pipe nozzles from outside the vessel and lifting the steam generator vertically.

d. Auxiliary Systems --

The major auxiliary systems necessary for safe and reliable reactor plant operation and the functions these

systems perform are as follows:

i) Emergency Cooling System -- prevents core damage by removing decay heat in one of two ways: 1) by using the steam generators either with coolant pumps operating or with natural circulation flow only, or 2) by using the purification system heat exchangers, bypassing the ion exchangers to get increased flow rate.

ii) Pressure Relief System -- prevents primary system overpressure by venting steam from the top of the reactor pressure vessel to the partially filled, 1000 gallon relief tank, where it is condensed; relief tank pressure in excess of 142 psi is relieved to the containment vessel; vented gases go from the relief tank to the gaseous waste disposal system.

iii) Primary Make-up System -- maintains sufficient water in the pressure vessel by replacing water lost due to buffer seal leakoff, coolant sampling, and system leakage; make-up water is pumped from the relief tank to the reactor vessel by one of two installed parallel-piston pumps.

iv) Purification System -- maintains primary chemistry within required ranges and reduces primary coolant activity levels by removing activated and not-yet-activated corrosion and wear products from the coolant; coolant is pumped through one of two regenerative heat exchangers and two mixed-bed ion exchangers by one of two canned motor pumps, to the buffer seal system.

v) Coolant Sampling System -- provides primary

coolant samples for analysis of chemistry and radioactivity levels.

vi) Intermediate Cooling System -- supplies fresh water cooling flow to various reactor system components such as primary coolant pump motor windings, purification system heat exchangers and neutron shield tank; the fresh water is in turn cooled by sea water.

vii) Ventilation and Air Conditioning System -- maintains required temperature and humidity conditions in the containment vessel (below 122F) and in the reactor auxiliary room; maintains a slight negative pressure in the containment vessel to enhance containment of any radioactive gaseous or particulate matter inside this vessel.

viii) Evacuation and Hydrogen Addition System -- prevents air entrainment when filling the primary system and effects recombination of oxygen produced by radiolytic dissociation of coolant in the core; also maintains a slight hydrogen overpressure in the steam dome in the reactor pressure vessel.

ix) Buffer Seal System -- supplies primary coolant from the purification system to the control rod drive buffer seals at 50 psig above primary pressure; leakoff which does not enter the pressure vessel through these seals is drained to the blowoff tank and recirculated by the primary make-up pump.

x) Liquid Waste Storage System -- collects and stores radioactive water for later dilution and overboard

discharge.

xi) Activated Water System -- stores and purifies primary coolant drained from the reactor for maintenance, in 3, shielded tanks of 1800 gallons each.

e. The Reactor --

i) The Core --

The reactor core makes up a right circular cylinder with active height of 44 in. and equivalent diameter of 45.2 in. As shown in Figure B-5, the core consists of 12 square and 4 triangular fuel elements. Each of the square elements is a 17 x 17 bundle with 226 fuel rods and 63 locations used for structural elements or burnable poison rods. All fuel rods are 0.429 in. OD, 0.0138 in. wall thickness, stainless steel tubes containing 0.40 in. OD sintered UO_2 pellets held in place by one end spring per rod. The burnable poison rods contain a mixture of ZrB_2 and ZrO_2. At their lower ends the rods are fastened to a tube sheet; their upper ends are positioned by pins in a top plate. The 2 end plates are connected by 1 central and 12 peripheral, square, zircaloy rods, the latter having contact surfaces against adjoining elements. Five intermediate grid plates with brazed ferrules are provided for additional support along the length of the fuel rods.

Each of the square fuel elements contains a control rod made up of 4, T-shaped sections guided in sets of U-shaped rails so that the center remains free for the central zircaloy structural rod. The absorbing section is made up of boron

carbide tubes banded together. Zircaloy followers on the lower ends of the rods prevent thermal flux peaking in a raised rod channel. The upper ends of each set of 4, T-shaped rods are connected together and coupled to a control rod drive mechanism shaft which penetrates the reactor pressure vessel top.

Fuel enrichment is varied in 4 radial zones to achieve sufficient flattening of neutron flux and more uniform fuel burnup. The average fuel enrichment is 4.02% U-235, with individual zone enrichments of 2.77, 3.20, 3.89 and 4.87% U-235. Core loading is 2.95 tons of UO_2. Average neutron flux in the core is 1.1×10^{13} n/cm^{2}sec. Average fuel burnup at the end of core life will be 7,200 MW days/ton uranium, corresponding to 500 full power days of operation. Total core heating surface area is approximately 1292 sq ft for the 3144 fuel rods.

The effect of the steam voids required in the core to maintain the pressure vessel steam dome is an overall loss of 0.6% Δ k, so that variations of reactivity and reactor power due to ship motion accelerations are small and require no compensating control rod motion. Maximum reactor accelerations measured in very heavy seas (wind velocities between 8 and 11 Beaufort) at full power were $\pm 0.2g$; resulting power level fluctuations were ± 1 MW as measured by special in-core neutron flux measuring instrumentation. Since this variation could not be detected on the steam generators, it has no effect on the main propulsion plant or turbogenerators. The

reactor design was based on maximum accelerations of $\pm 0.5g$, as required by the rules. Relatively constant primary pressure is maintained by operating the reactor in a constant core outlet temperature mode; due to the negative temperature coefficient tending to maintain constant average temperature, control rod motion is required for power changes.

ii) The Reactor Pressure Vessel --

The 7 ft 6 1/2 in. ID, 28 ft 6 in. high, 72 ton reactor pressure vessel, shown in Figure B-4, is a vertical cylinder constructed of 2 seamless, hollow-forged hoops with forged, hemispherical ends; all parts are made of fine grain 15 MnMoNiV53 (Society Stahl und Röhrenwerke Reinsholz) steel. Penetrations through the vessel include: 3 primary coolant pump nozzles in the lower head; 12 control rod drive nozzles and piping nozzles for pressure relief valves and sensing lines in the upper head; and 3 feedwater and 3 steam piping nozzles, each with built-in radiation absorption plugs, in the upper part of the cylindrical portion of the vessel. Internal vessel surfaces are double-layer weld-deposit clad with 0.334 in. of 18/9 CrNi steel for corrosion protection. The vessel is designed for a pressure of 1250 psi at a temperature of 572F; wall thickness is 2.3 in. The upper head is flanged and bolted on with 36 bolts of 0.4 in. diameter; the flanged joint is sealed with 2, silvered austenite O-rings. The reactor vessel is supported by the containment vessel via a foundation of 12 radial brackets similar to the 24 brackets with which the ship structure supports the con-

tainment vessel.

f. Control Rod Drive System --

The 12 control rod drive units are of the rack and pinion type with a rotating buffer seal and a normal driving speed of 5 in./min. These units are mounted on a common platform above the reactor vessel and consist of electric motors equipped with reduction gears coupled to the rack and pinion through a Cardan Coupling, scram springs, and electromagnetic scram-couplings. Rods can be operated individually or in groups, manually or automatically. The scram springs insert the rods in less than 5 sec to 2/3 of full stroke with a 45° ship list. Three capacitive level gauges in a water-filled U-tube transmit list and heel signals to the reactor safety which automatically scrams the reactor for ship trim in excess of 12° or heel in excess of 45°. Two out of 3 coincidence circuitry reduces the chance of spurious scrams from this system and permits maintenance at power.

4. POWER PLANT DESCRIPTION; PROPULSION SYSTEM --

a. The Main Propulsion Unit --

The main propulsion unit is a conventional, double-reduction, articulated gear, high-(HP) and low-pressure (LP) turbine set: a moisture separator is in the line between HP and LP turbines. Since the incoming HP steam is slightly superheated, only the LP turbine has increased draining provisions in the lower pressure stages. Steam from the LP turbine exhausts to the main condenser, from which it is pumped

to the deaerator for deaeration at 46 psig, 270F; one of two turbo feed pumps or the backup electric feed pump returns the feedwater to the steam generator. The all-impulse HP turbine consists of 1 Curtiss wheel followed by 5 impulse stages, and is designed for 6,050 rpm; design speed for the 4 bladed propeller is 97 rpm. The single-flow, all-impulse LP turbine has 6 impulse stages and is designed for 3,185 rpm; a Curtiss wheel on the aft end of the LP rotor provides 4,000 SHP at 47 shaft rpm for astern operation. The main condenser is a 2-pass unit designed for 95% vacuum and 14,250 gpm seawater flow rate. Figure B-6 provides a heat balance for the plant.

b. The Electrical System --

Electrical power is normally supplied by one of two 450 kw, 380 volt, 50 Hz, geared, turbo generators. These sets normally exhaust to the main condenser at sea and to an auxiliary condenser in port. A 450 kw, diesel generator, located in the auxiliary engine room, is provided as a backup for the turbo generators, and an emergency diesel generator of 240 kw is located on the boat deck. The auxiliary generator can feed all electrical systems on the ship and can supply enough electrical power to feed all normal loads, both conventional and nuclear. The emergency generator can supply enough electrical power to start the reactor from cold. Both the auxiliary and the emergency generators automatically pick up the load on the emergency switchboard within 7 to 30 seconds following loss of output from the turbo generators; this load consists of all vital reactor plant loads plus those required

for ship navigation. Reactor instrumentation loads are carried by a special stabilized grid fed from the emergency switchboard by converters and buffered by a battery.

5. REACTOR SERVICE ROOM --

To facilitate research and make the ship independent of shore facilities, a reactor service room, shown in Figure B-3b, is installed on board. A 35-ton crane, located on the main deck, can handle the lead-shielded, fuel element transfer cask stored in this room, containment and pressure vessel covers, and other parts as necessary in the course of refueling. A 3-ton, overhead travelling crane facilitates handling of tools and other small loads in this room. A water-filled, water-cooled, heavy-concrete shielded, stainless steel lined, storage basin is provided to contain all expended fuel elements in boxes made of boron-containing materials. This basin is approximately 4 ft 3 in. wide, 21 ft 2 in. long and 26 ft 5 in. deep. Intended storage periods for expended fuel elements are in excess of 100 days to preclude the necessity for water cooling of containers used to transport the elements from the ship for processing and disposal. GKSS does not intend to include such a facility in future nuclear merchant ships, since complete fuel cycle service has been developed for land-based reactors and can be used for refuelling shipboard reactors.

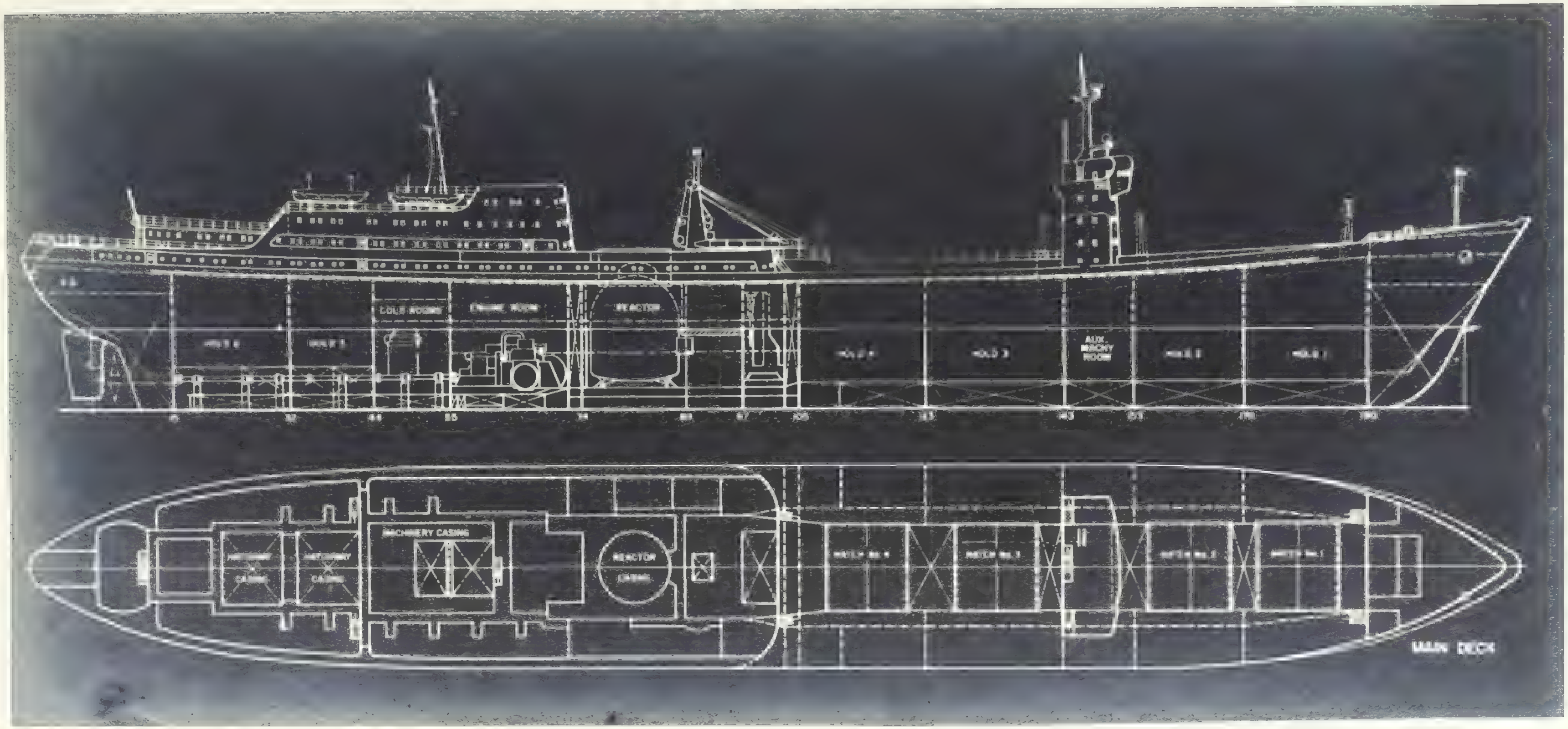


Figure B-1. N.S. OTTO HAHN General Arrangement

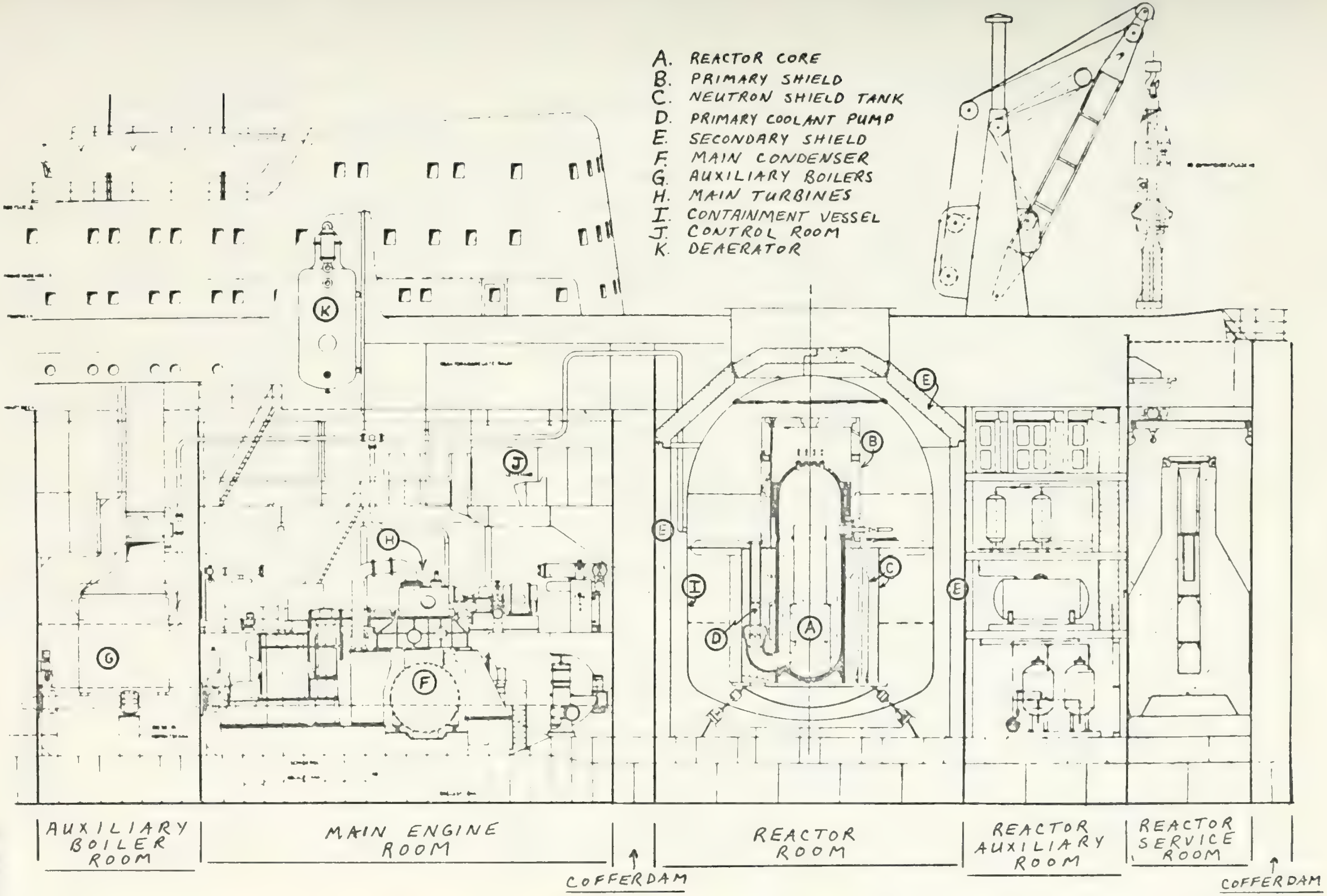


Figure B-2. N.S. OTTO HAHN Nuclear Power Plant Arrangement

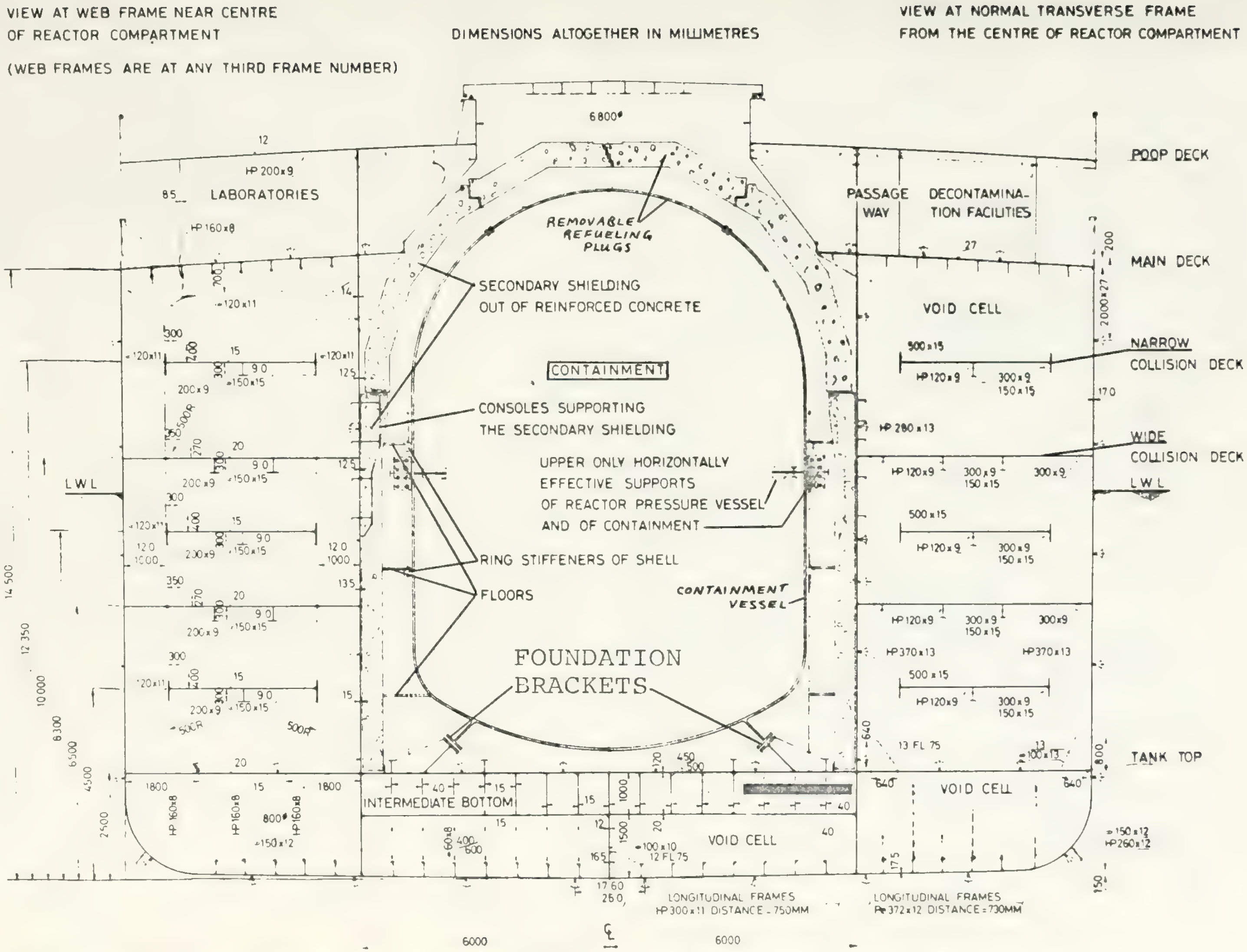
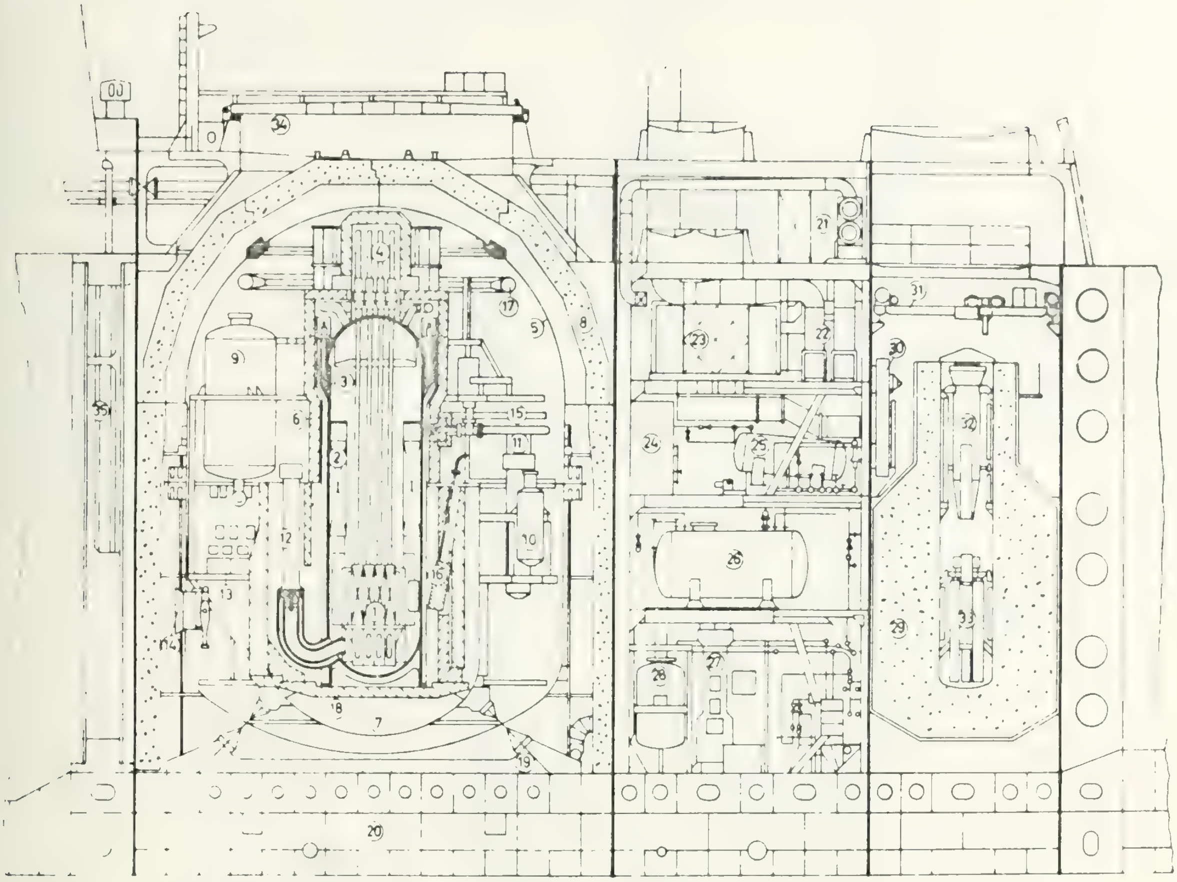


Figure B-3a. Details of N.S. OTTO HAHN Reactor Area Structure



1. Reactor Core
2. Steam Generator
3. Pressure Vessel
4. Control Rod Drive Mechanisms
5. Containment Vessel
6. Primary Shielding
7. Neutron Shield Tank
8. Secondary Shielding
9. Blowoff Tank
10. Purification System Ion Exchanger
11. Purification System Heat Exchanger
12. Primary Coolant Pump
13. Primary Makeup Pump
14. Flooding Valve
15. Steam & Feedwater Piping
16. Neutron Detector
17. Ventilation Ducting
18. Support Plate
19. Containment Vessel Fdn.
20. Double Bottom
21. Ventilation System Supply
22. Ventilation System Return
23. Active Carbon Filters
24. Sampling System
25. Waste Gas System
26. Active Water Storage Tank
27. Ion Exchanger
28. Sampling Tank
29. Service Pool
30. Rotating Cover
31. Service Crane
32. Lead Cask
33. Fuel Element Rack
34. Reactor Access Hatch
35. Wireway

//// Cast Iron
Concrete

Figure B-3b. N.S. OTTO HAHN Reactor Plant

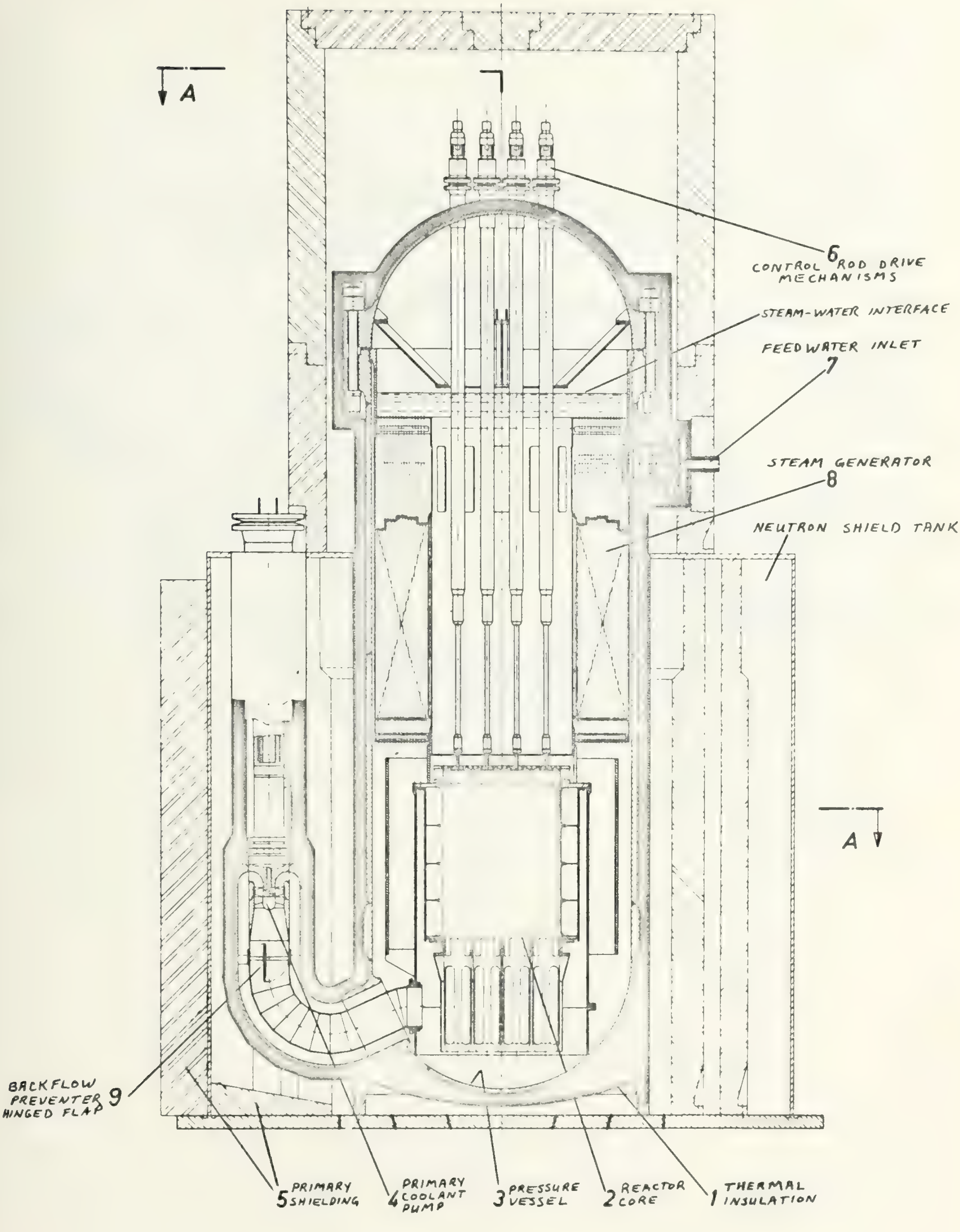
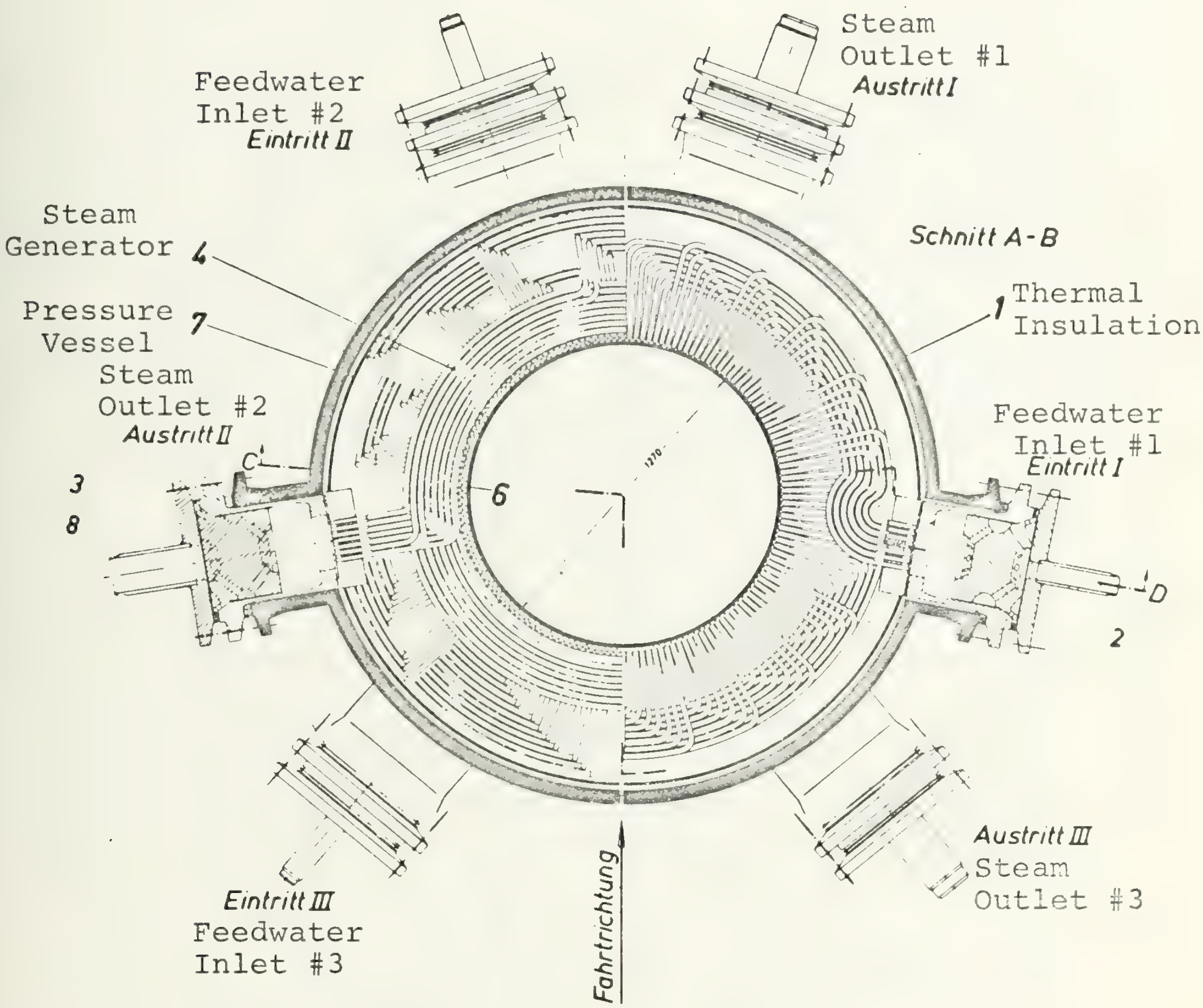
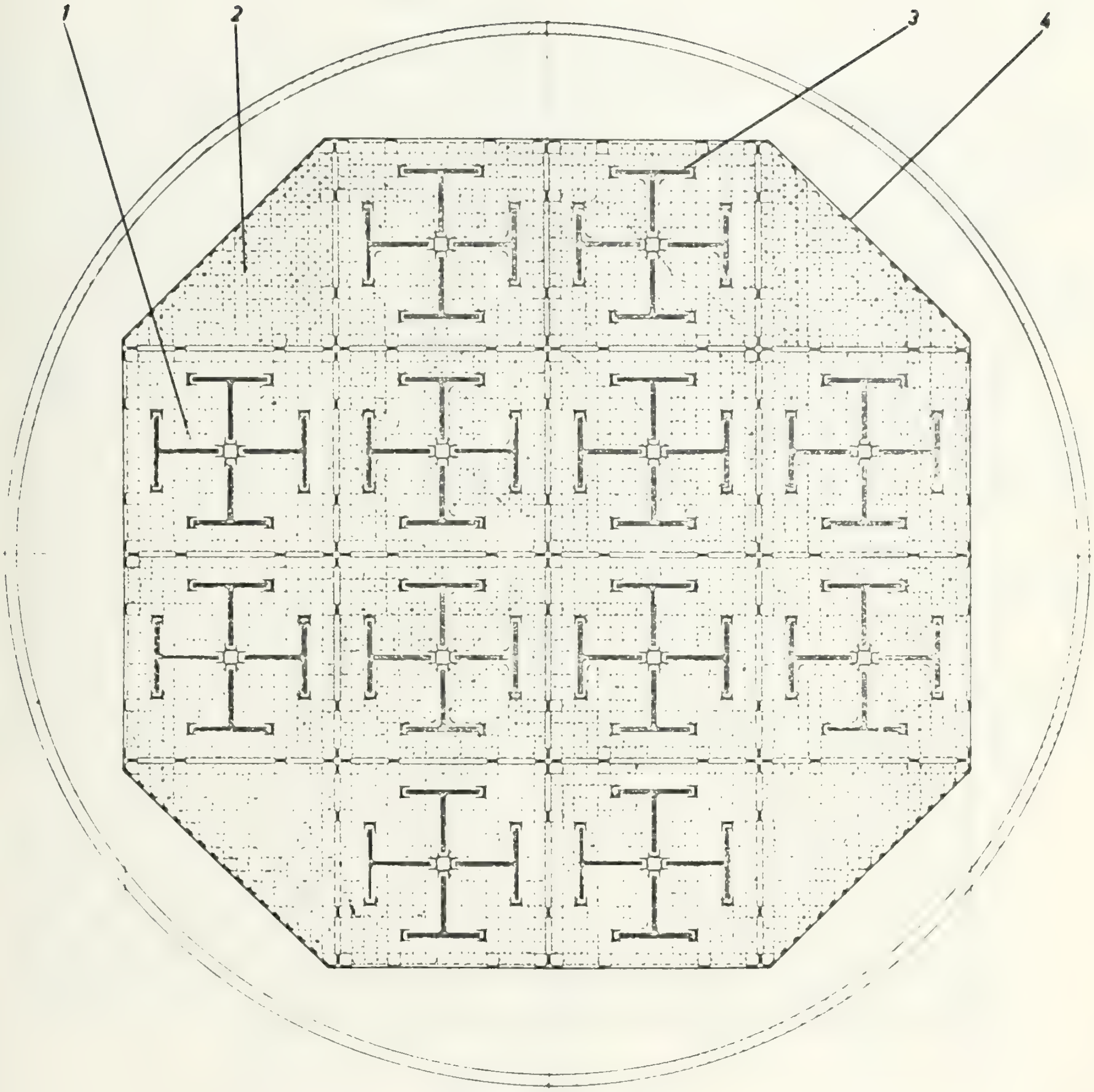


Figure B-4a. N.S. OTTO HAHN Reactor Complex



PLAN VIEW

Figure B-4b. N.S. OTTO HAHN Steam Generator

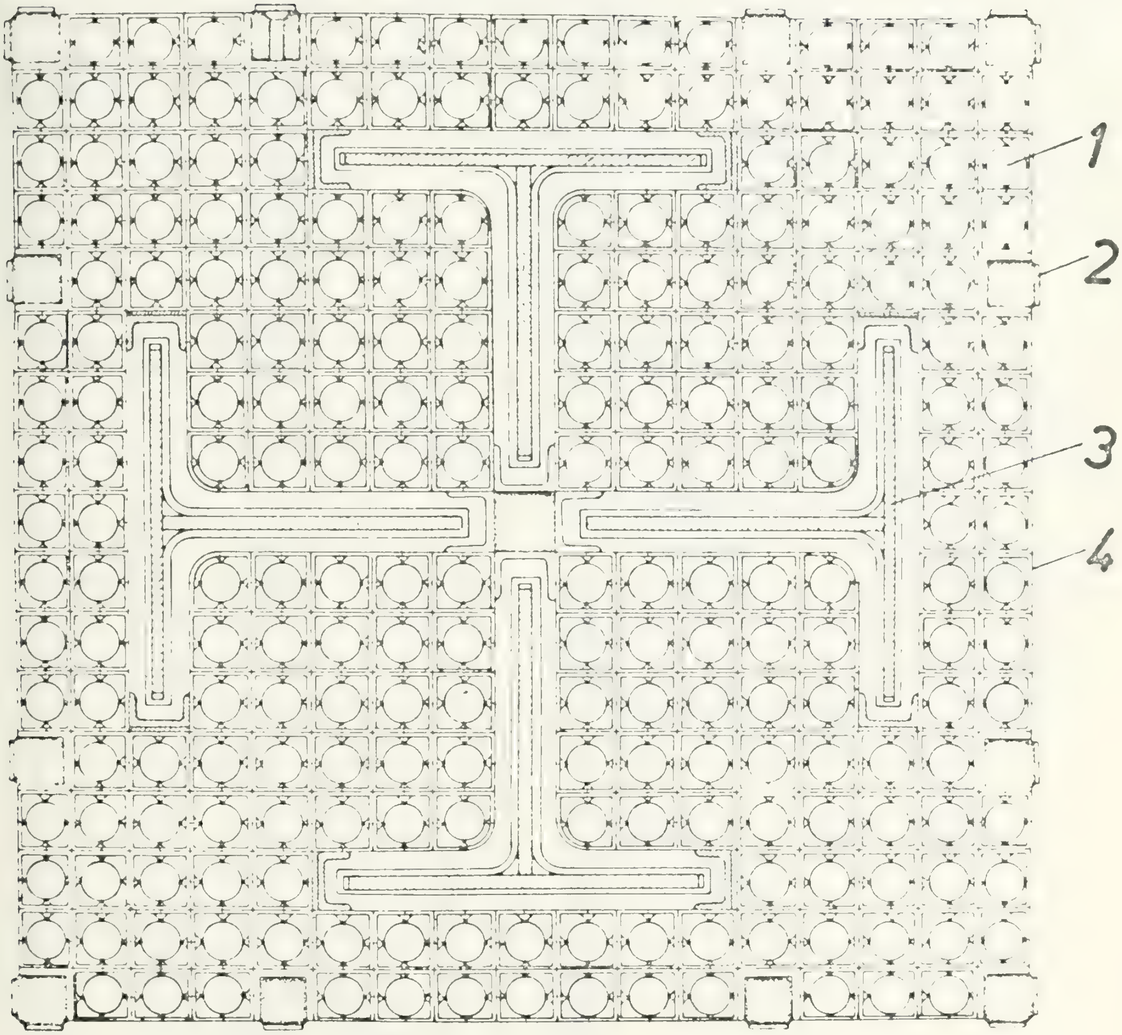


PLAN VIEW

1. Square Fuel Element
2. Triangular Corner Fuel Element
3. Control Rod
4. Core Support Shell

Figure B-5a. N.S. OTTO HAHN Core Cross Section

Schnitt C-D



1. Fuel Rod
2. Structural Rod
3. Control Rod
4. Fuel Rod Support Grid

Figure B-5b. Square Fuel Element Cross Section
N. S. OTTO HAHN

00237

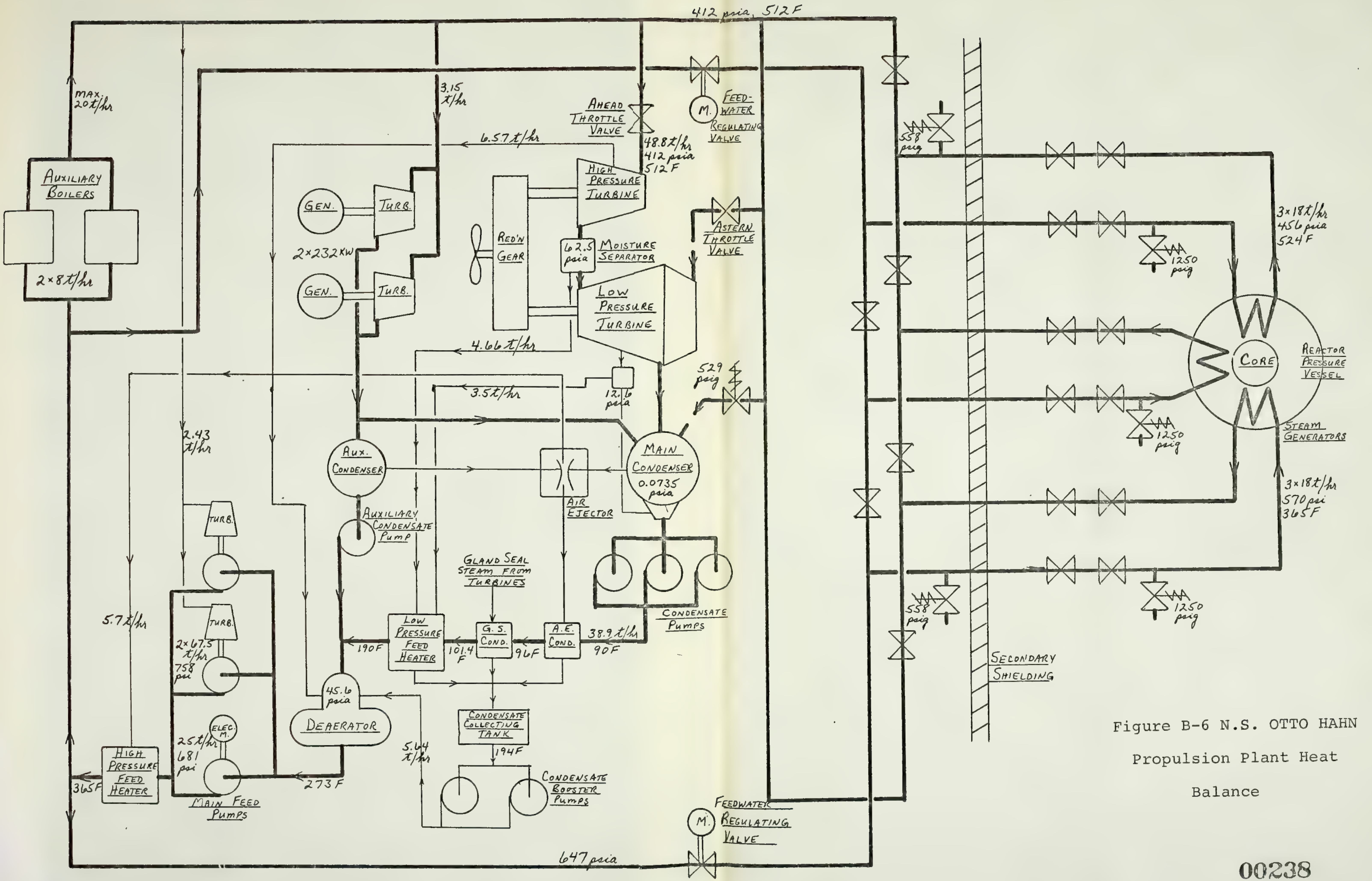


Figure B-6 N.S. OTTO HAHN Propulsion Plant Heat Balance

00238

C. N.S. LENIN (ref's. 18, 44, 55 through 59)

1. GENERAL --

LENIN is the world's first nuclear-powered non-naval ship. Designed and built by the U.S.S.R. to advance the economic development of the Soviet Northern Regions, LENIN is a 16,000 ton, turboelectric-drive icebreaker. With a total power of 44,000 SHP divided among 3 shafts, LENIN was the world's most powerful icebreaker when she was built. The use of nuclear propulsion for an icebreaker resulted in greatly improved operational capabilities; examples of this are: 1) greater installed power provides greater open water speed and increased icebreaking capability, and 2) freedom from the frequent bunkering required by conventional icebreakers affords longer ranges in any desired zone of operation, much more prolonged periods on station, unrestrained use of high power for any desired purpose, and no lack of auxiliary power should the vessel become icebound.

To ensure that continuous propulsion and auxiliary power would be available, LENIN has 3 reactors of the pressurized light water type, any 2 of which can provide full power. The third reactor is normally operated only in heavy ice navigation; in case of loss of power from 1 reactor, the ship thereby loses neither speed nor capacity. In addition, the reserve reactor facilitates routine inspection and repairs at power, as well as making it possible to reduce considerably the excess reactivity required to compensate for xenon poisoning, thereby permitting either increased fuel burnup or use

of lower thermal neutron fluxes and more stable heat release rates.

Construction of LENIN at the Kirov Elektrosia Works in Leningrad was begun August 25, 1956, and launching took place December 5, 1957; she was commissioned and joined the Arctic fleet December 3, 1959, with a specific capacity of 2.75 SHP/ton of displacement, almost 1.5 times that of any other icebreaker then in operation. Other specifications of LENIN are as follows:

Length Over All	440 ft
Beam	90 ft 6 in.
Draft	29 ft 10 in.
Maximum speed in open water	18 knots
Cruising speed in pack ice	3-4 knots
Screw thrust, bollard pull conditions	330 tons
Propulsion plant weight, total	5,767 tons
Nuclear plant, including radiation shielding	3,017 tons
Machinery plant other than nuclear	2,750 tons
Complement	230 men

Ability to navigate in ice fields is increased by a relatively high beam-to-length ratio and by a special heeling and trimming system using automatic and centrally controlled, reversible propeller electric pumps, each with a flow rate of 66 tons per minute. No bow screw is installed.

2. SHIP ARRANGEMENT AND STRUCTURE --

LENIN is divided into 12 watertight compartments by 11 main transverse bulkheads; any two of these compartments may be flooded without loss of the ship. Designed to withstand any compression expected from ice jams under Arctic conditions, the hull is constructed of a grade of highly resistant steel with good crack arrest properties at low temperatures; hull plating is 2.05 in. thick in the bow section, 1.42 in. thick in the middle, and 1.73 in. thick in the stern. To reduce open water rolling accelerations to below that of most other icebreakers, the transverse metacentric height was designed to be 6 ft 3 in. to give a rolling period of not less than 10 sec.

Two longitudinal bulkheads (situated at one-fifth the beam from the ship's sides at the reactor compartment) extend from the double bottom to the upper deck; these bulkheads form, on each side of the ship, voids for heeling, ballast, fuel, and other tanks (below the lower deck) and spaces for various laboratories, service rooms and personnel cabins (above the lower deck). Machinery plant rooms are accommodated by the space formed between these two bulkheads. The safety of this subdivision scheme was demonstrated October 12, 1961, when the LENIN struck a sunken iceberg at 17 knots and continued normal operations for 6 weeks with outer compartments flooded between transverse bulkheads 36 and 48.

The machinery arrangement, shown in Figure C-1, consists of 4 basic compartments: a reactor compartment

amidships; two turbogenerator rooms, one forward and one aft of the reactor compartment; and a propulsion motor room. To accommodate the large machinery plant in the relatively short hull length, maximum use was made of "stacking" of equipment. Use of electric propulsion, in addition to enhancing maneuverability, allowed the 4 main turbogenerators and their condensers and air ejectors to be mounted high in the ship: in the forward engine room, above the auxiliary turbogenerators, other auxiliary machinery and the distilling plants; and in the after engine room, above the two outboard ("wing") shaft motors and their maintenance mechanisms and equipment. The reactor compartment contains the 3 reactors and the 3 primary systems and is enclosed by the secondary radiation shield/ containment vessel complex.

The flush deck with moderate shear provides mounting for a prolonged superstructure, storage of launches and lifeboats, and a hangar and landing platform for 2 reconnaissance and communication helicopters in the stern section. A 40-ton winch is provided for ship towing, and 3 electric cranes for freight operations. To permit extended independent operations, the ship is equipped with a medical unit comprising complete facilities for surgical, physiotherapeutic, dental and x-ray services, plus a sick bay, a pharmacy and an isolation ward.

3. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

The 3 pressurized light water reactor systems in LENIN are independent of each other and can supply a total of 360 tons of steam per hour. Except where otherwise indicated,

the descriptions below apply to any 1 of the 3 reactor systems.

a. Containment Vessel --

The LENIN design does not provide for a containment vessel as such. However, containment of fission products in the event of a collision of grounding is enhanced by the unusually large (compared with other ship types) degree of structural strength built into icebreakers to preclude ice damage, and by the approximately 14 in. thick steel secondary shield which surrounds the primary system (see Figure C-1).

b. Radiation Shielding --

The 1,963 ton radiation shield consists of primary and secondary shielding. Primary shielding consists of a shield tank concentric with and enclosing all but the upper portion of the reactor vessel; this tank contains equal volumes of stainless steel and water arranged in layers to reduce total shield weight. Total primary shield steel thickness is 22.5 in.; total water thickness is 44.5 in. In addition to this installed shielding, the thermal shields, internal reactor vessel structure and reflector provide 8.9 in. thickness of water and stainless steel which attenuate radiation from the core. The tank is sectioned to minimize shielding loss in case of a leak, and is fitted with cooling coils served by water from an auxiliary cooling system.

Because of the complex arrangement of piping and equipment in the vicinity of the reactor vessel top, this area is shielded with heavy limonite concrete. In addition to this

installed shielding, the internal reactor vessel structure and reflector provide the following which attenuate radiation from the core: below the core, 35.5 in. steel and 29.5 in. water; above the core, 38.2 in. steel, 19.7 in. water and 27.5 in. heat-resisting, graphite-based compound.

Secondary shielding consists of steel walls 12 to 16 1/2 in. thick surrounding the primary system. The shielding is designed to keep radiation levels 9.8 ft from the secondary shield less than 0.07 to 0.18 mrem/hr, 0.1 to 0.3 of permissible dose for an 8-hour working day.

c. Primary System --

Each reactor system has a two-loop primary piping system to circulate primary coolant. As shown in Figure C-2, each loop consists of 4 electrically actuated, gate type, loop isolation valves, 1 steam generator, 2 main and 1 emergency primary coolant pumps, and connecting piping. The light water primary coolant is maintained at a pressure of 2950 psia by the pressurizing system. All primary system components are designed for a pressure of 3300 psia at operating temperatures. Each steam generator at full power produces 120,000 lb/hr total of 142.5F superheated steam at a pressure of 412 psia and a temperature of 590F. Reactor outlet temperature is 603F; inlet temperature is 502F. Primary coolant flow rate through the core is 3680 gpm. Isolation of one loop results in a maximum power level for that reactor of 55%.

The 1840 gpm at a head of 330 ft of water, main primary coolant pumps each have a closed type centrifugal

impeller with a one-sided axial inlet; the 250 kw pump motors are 3 phase, 380 volts, ac, with canned squirrel cage rotors and nichrome-canned stators. The 2 main pumps in each loop are powered from different electrical sources to enhance plant reliability and reduce the possibility of fuel cladding meltdown due to the high temperatures and heat content and the low thermal conductivity of the dioxide fuel following rapid reduction or loss of primary coolant flow. Labyrinth gaskets on both sides of the impeller relieve axial stresses; axial rotor forces are resisted by a thrust pin on the rotor shaft. The cast pump helix is welded to the high strength outer casing. Cooling for the 2 stellite-lined journal bearings and for the motor windings is provided by filtered primary coolant flowing in a pump inner cooling circuit and by auxiliary cooling system water flowing in a pump outer cooling circuit. This pump and the emergency pump are shown in Figure C-3. The construction of the emergency pump is similar to that of the main pump except that a special plastic material is used for the bearings; the emergency pump starts automatically upon loss of power to both main pumps in its loop, in order to secure removal of decay heat from the core.

Steam generators are of a uniflow boiler type with a vertical cylindrical steel reservoir, whose piping system consists of loosely defined zones for economizer, vaporization and steam superheating. Each steam generator has a total surface area for heat transfer of 4,035 ft^2. Steam pressure is kept constant over all power levels by means of 2 inter-

connected systems: one for reactivity control, and one to control the heat transfer rate from the primary coolant to the steam. Reactivity control is effected by the control rods, and heat transfer rate is controlled by actuation of a valve regulating the rate of feedwater flow to the steam generators; a thermocouple measuring reactor inlet temperature provides a basic input for this system. Variation of feedwater flow controls steam pressure by shifting the interfaces between, and thereby the amount of heat transferred in, the economizer, vaporization and superheating zones of the steam generator.

d. Auxiliary Systems --

i) Pressurizing and Volume Compensating System --

Connected to the inlet piping of each reactor is a set of 4 pressurizers which maintain primary coolant pressure within a 75 psig band and provide a surge tank to accommodate primary coolant expansion and contraction without formation of a steam void in the rest of the primary system. Each pressurizer is a high pressure vessel containing volumes of water and steam; pressure is controlled by the output of replaceable electric heaters in the lower portions of these vessels. Sufficient water volume is provided to allow complete primary system cooldown without any additional supply of water into the system.

ii) Purification System --

Each loop is provided with an ion exchange filter in a piping loop across the primary coolant pumps, and

a cooler connected in series with this filter. This system decreases the fouling of heat transfer surfaces and equipment and reduces the level of radioactivity in the primary coolant by maintaining primary system pH between 6.0 and 8.0 and Cl^- content below 0.02 mg/l, and by removing other dissolved ions from the coolant. Filter flow rate is approximately 18 gpm, 1% of the loop flow rate, and is used to cool the primary coolant pumps before being returned to the loop. Filter inlet temperature is 95F.

iii) Auxiliary Cooling System --

Two auxiliary cooling systems are provided: 1) an interior (within the secondary shield) system to cool the primary shield tank (this tank is cooled by a separate, closed system for positive isolation of its activated corrosion products); and 2) an exterior (outside the secondary shield) system to cool purification system coolers, primary coolant pumps, and the interior cooling system. The exterior system is cooled by sea water.

iv) Radioactive Waste Collection, Storage and Processing System --

This system provides a means for piping radioactive water into shielded 800, 2500, and 6600 gal tanks and for reduction of its activity level to 5×10^{-9} curies/l by use of special filters.

v) Radioactive Ventilation System --

Each of the spaces housing a reactor plant is maintained at a reduced air pressure to prevent outleakage

of airborne radioactivity. Exhaust air from these spaces is filtered to remove radioactive particulate and pumped overboard through the hollow main mast at a rate of 2,805,000 ft^3/hr. The activity of this exhaust air is normally below 2×10^{-10} curies/l, the minimum activity level detectable by on-board instruments.

e. The Reactor --

i) The Core --

The reactor core has an active equivalent diameter of approximately 3.3 ft and an active height of approximately 5.2 ft. Core height-to-diameter ratio was made high to facilitate overriding of xenon poisoning and to provide a greater fuel burnup rate. In the 8.9 in. annulus between the core and the reactor vessel inner wall are mounted several concentric steel cylinders which, with the intervening coolant, serve as thermal shields to protect the reactor vessel from the intense neutron and gamma radiation emanating from the core. Primary coolant enters the central fuel element flow channels from the bottom, flows upward and leaves at the top; there it is redirected to flow down along the shields and enter the peripheral fuel element flow channels at the bottom. Flowing upward through these channels, the coolant leaves the vessel through the outlet piping.

Each of the 219 fuel elements consists of 36 zirconium alloy, 0.24 in. OD, 0.0295 in. thick tubes into which are loaded sintered uranium dioxide pellets with annular space provided to accommodate fuel expansion. These fuel rods are

mounted within 2.6 in. diameter, cylindrical, stainless steel shroud (the flow channel) in a triangular lattice arrangement with a 2.52 in. pitch such that they form concentric circles with 6, 12 and 18 rods.

The choice of dioxide fuel over metallic was determined by engineering considerations: chemical stability in water, stability under irradiation, absence of allotropic transitions up to the temperature of fusion, and easier decontamination of the primary system in the event of fuel cladding failure. Fuel enrichment is 5% U-235; initial core loading is 85 kg U-235. The fuel in the central fuel elements contains an admixture of 92 g total of natural boron isotopes to reduce initial excess reactivity and flatten the radial thermal neutron flux distribution, thereby providing more uniform heat production and fuel burnup.

Each reactor has a maximum thermal capacity of 90MW and is designed to operate at a nominal full power of 65MW for 1 year. Maximum fuel element surface heat flux is approximately 205,000 Btu/ft^{2}hr. Refueling may be accomplished either element-by-element or by replacing the core as a whole. The core was designed to have a moderate value of negative temperature coefficient (NTC) over normal operating temperatures; the value chosen was a compromise between the requirement to have large control rods to compensate for excess reactivity as temperature is reduced with a large NTC, and the requirement to have high coolant flow rates to minimize water temperature fluctuations in plant power

transients with a small NTC. The average fuel burnup is 13,000 MWd/ton; maximum is 30,000 MWd/ton.

ii) The Reactor Pressure Vessel --

The reactor pressure vessel is a 6 ft 1 in. OD, 15 ft 3 in. high, vertical cylinder with an integral bottom and a self-sealing upper cover. The vessel is constructed of a low alloy, high strength carbon steel and is protected from high temperature water corrosion by a stainless steel shell insert. The vessel is shown in Figure C-4. The bottom of the vessel contains the single primary coolant inlet pipe, and the upper portion of the vessel contains the 2 fore-and-aft-oriented, coolant outlet pipes. The bolted-on upper cover contains the penetrations necessary to accommodate the control rod drive mechanisms.

f. Reactor Control System --

The reactor is controlled from the main control room, in which the main instrument panelboard is situated. Two automatic regulators are provided; one is an installed spare. Each automatic regulator controls the degree of insertion of 3 regulating control rods and is designed to handle all short-term reactivity transients; each group of 3 regulating rods has an average worth in the clean core of 0.36% $\Delta k/k$.

Slow reactivity changes due to fuel burnup, fission product poisoning and deposits on heat transfer surfaces are compensated for by a set of special control rods designed to be semi-transparent to neutrons to decrease their effect on

core neutron and heat flux distributions; these rods are driven by a single shaft, with a screw thread and ball nut mechanism that are motor driven through reduction gearing. No separate rods are provided for manual control; the reserve regulating rods are ordinarily used for that purpose. Under emergency conditions the reactor is scrammed by dropping spring-driven safety rods; time to full insertion of these rods is 0.6 sec.

g. Control Rod Drive System --

There are 3 distinct types of control rod drive systems provided. Figure C-5 shows the 2 automatic regulators and the compensating control rod drive unit; for simplicity, only 1 of the 3 regulating rods driven by each circular rack is shown in this figure. The safety control rod drive system consists of electric motors driving gears meshed with racks on the rod shaft extensions. Between the motors and the gears are mounted electromagnetic couplings. Heavy springs are compressed as these rods are raised; a scram signal deenergizes these couplings, allowing gravity and spring force to drive the rods into the core at high speed. Electromechanical relays used in the original design of the control system were replaced in 1963 with semiconductor devices to obtain greater reliability. At this same time certain of the scram functions were changed to power cutbacks (to below 30% power) to provide increased reactor availability and reduction in the number of sudden temperature changes the reactor will be subjected to.

4. POWER PLANT DESCRIPTION; PROPULSION SYSTEM --

a. The Main Propulsion Unit --

The 4 single-casing main turbines, operating at full power inlet steam conditions of 412 psia, 590F, 142.5F superheat, each drive 2 double armature electric generators via a single stage reduction gear. Although less efficient than a double-casing turbine, the single-casing unit permits elimination of the intermediate reduction gear and thus a decrease in the number of bearings and gaskets; assembly is thereby simplified and operation made more reliable. Speed is controlled by inlet steam throttling. To provide the ruggedness needed for frequently and abruptly changing loads, reactive blading was selected for its larger axial clearances and simplicity. An automatic device is installed for each main turbine unit to pass excess inlet steam directly to the main condensers during rapid down-power transients in order to secure smoother steam generating plant operating conditions.

Continuous flow of seawater (8 times that required for a diesel plant of the same horsepower is needed) is ensured by installation of 2 separate, side intake sea suction boxes for each condenser; each box has interior and exterior protective screens to minimize the probability of ice choking the flow path. The boxes are interconnected by channels in the ship's double bottom; each box has sufficient capacity to individually supply its condenser. The mechanical plant can be controlled from local panelboards and remotely from the propulsion plant control desk in the main control room.

The propulsion system is shown schematically in Figure C-6. For increased plant reliability all main units such as seawater, condensate, feedwater and lubricating oil pumps and the main turbogenerator seawater circulation and condensate pumps are driven by steam turbines for reliability; all other auxiliary pumps are electric motor driven. The parallel-connected feedwater pumps are provided with automatic controls to maintain constant pressure drop across the feed valve; if one pump should stop, the other automatically picks up speed to maintain an uninterrupted supply of feedwater to the steam generators. The electric, reserve, main turbine lubricating oil pumps are automatically started when oil pressure drops to a preset limit.

All main steam, condensate and feed piping is in the shape of ring mains linking the 2 engine rooms and ensuring delivery of the working fluid reliably along either side for uninterrupted flow under any casualty condition. All condensate and associated heat exchange equipment is made with double tube sheets to minimize the likelihood of seawater contamination; all makeup water is doubly distilled and ion exchange filtered.

Non-propulsion plant steam (e.g. for space heating, laundry, showers, etc.) is generated in special boilers heated by propulsion plant steam; this feature minimizes spread of radioactivity in the event of primary-to-secondary system leakage in the reactor system steam generators. Auxiliary steam at a rate of 22,400 lb/hr can also be supplied, by 2

oil-heated water-tube boilers, in the event all reactors are secured. Because of the frequent load variations associated with icebreaking, a single stage feedwater heater heated by exhaust steam from the auxiliary turbines was selected; this exhaust steam, at a pressure of 30 psia, is also the heat source for the distilling plants. Feedwater is heated to about 212F.

b. The Electrical System --

The electrical propulsion system is shown schematically in Figure C-7. The 3 propulsion motors, with shaft horsepowers distributed in the ratio 1:2:1, are each driven by 4 generators in series/parallel combinations. The 1:2:1 power ratio places half the available power on the center shaft, the least vulnerable to ice damage, and leaves 75% power still available in the event one of the wing shafts becomes inoperable due to blade damage of loss. For maximum controllability of the propulsion motors,dc machines were selected. Each motor is an enclosed, double armature unit with a closed air cooling cycle and forced lubrication bearings; armature voltage is 1200 volts.

The center shaft motor has a continuous capacity of 19,600 hp (9,800 hp per armature) and each wing shaft motor has a capacity of 9,800 hp (4,900 hp per armature). Each motor has 3 excitation units, two working and one reserve. Each excitation unit consists of 4 separate components: an ac driving motor; a constant-voltage generator for control circuit supply; a double stage, high gain amplidyne exciter

for 1 armature of the propulsion motor; and a double stage, high gain amplidyne exciter for 2 of the electrical generators driving that motor.

The dc propulsion generators are 1200 volt, double armature units with a capacity per armature of 1,920 kw at 600 volts and 595 rpm; closed cycle air coolers are used. The 8 generators are installed in 4 pairs, each pair driven by a steam turbine. Each pair of generators consists of 1 generator with parallel-connected armatures comprising a single, 3840 kw, 600 volt unit, and 1 generator with 2 independent, 1920 kw, 600 volt armatures. Each pair of generators thereby supplies power to all 3 propulsion motors simultaneously. Via complex switching gear allowing all possible combinations to be used, each of the 1200 volt armatures of each propulsion motor is driven by two 600 volt generator armatures connected in series, as schematically shown in Figure C-8.

Control of the propulsion motors and the entire electrical plant is provided by the propulsion plant control desk in the main control room, with propulsion motor remote control stations located in the wheelhouse and on the bridge. The use of amplidyne exciters secures smooth starting and reversing, makes it possible to have electric motors under current in case a screw is stuck, and limits the recuperation of power from screws to turbines in reversal in free water. The small capacity required for control permits compact selsyns to be used at the control panels.

The 2 auxiliary turbogenerators produce 6,200 kw, 3 phase, 50 Hertz, ac electricity at 380 volts. This power is used to drive the short-circuited, asynchronous motors used for ship auxiliaries and 127 volt transformers for lighting and general purpose uses; these ac motors are simpler, cheaper to make and operate, and more reliable than comparable dc motors.

5. NUCLEAR SERVICING VESSEL, LEPSE --

Refueling of LENIN is accomplished with the aid of the nuclear servicing ship, LEPSE. This ship is equipped with spent fuel storage and other facilities, including a 12-ton crane. Fuel elements are removed from the reactor into a shielded flask, fitted with a mechanism for lifting the fuel elements into the flask, and transferred in this flask to the storage facility on the LEPSE for decay and disposal.

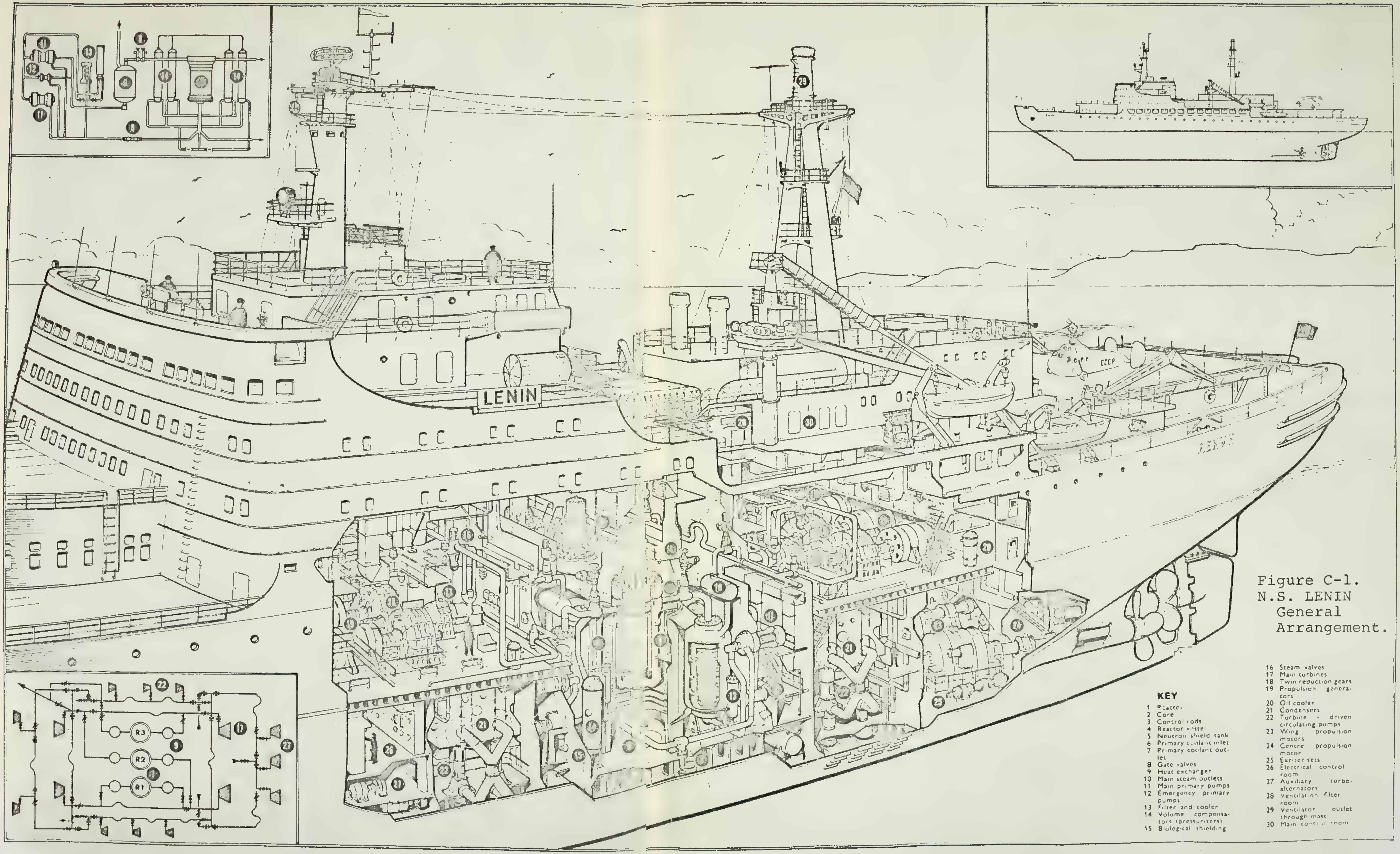
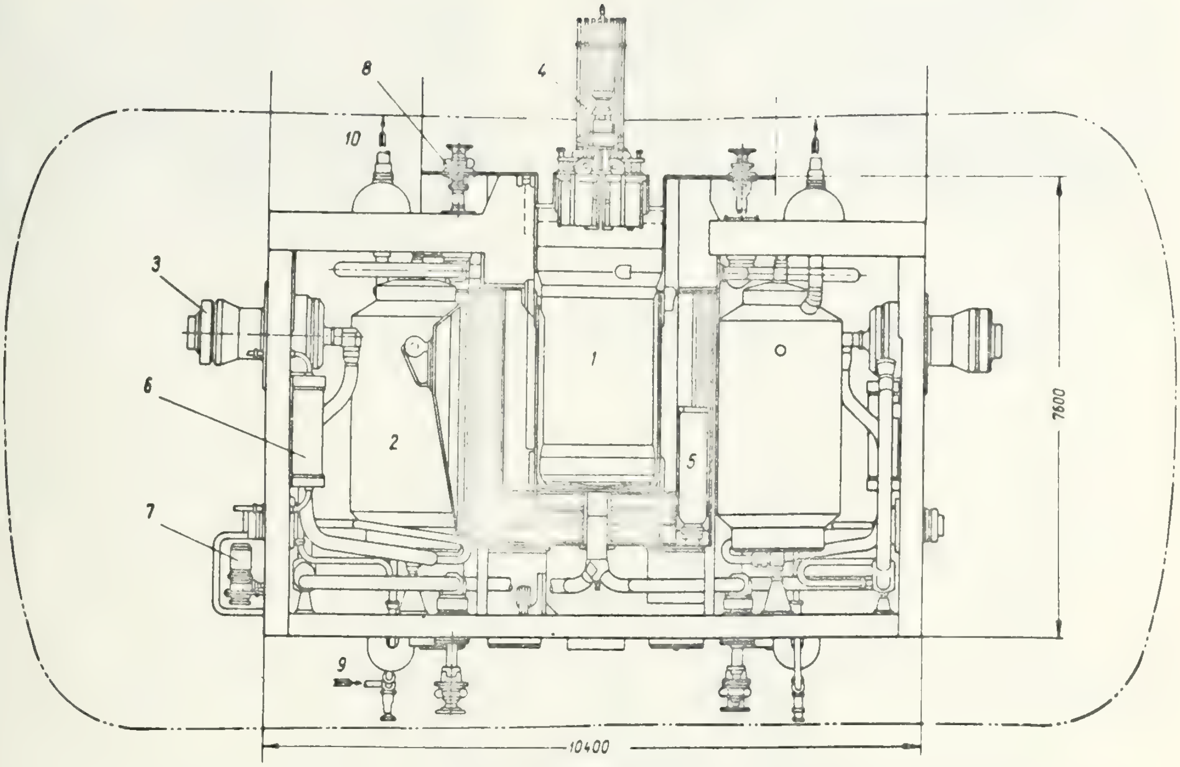
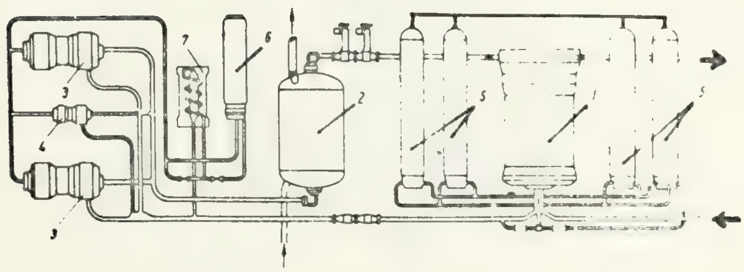


Figure C-1.
N.S. LENIN
General
Arrangement.



General layout of the steam generator portion

1, Reactor; 2, steam generator; 3, main circulating pump; 4, safety and control system mechanism; 5, filter; 6, cooler; 7, inner circuit pump; 8, primary circuit valve; 9, feedwater inlet; 10, steam outlet



Principal scheme of the steam generator portion

1, Nuclear reactor; 2, steam generator; 3, main circulating pump; 4, emergency pump; 5, volume compensators; 6, filter; 7, filter cooler

Figure C-2 N.S. LENIN Primary System

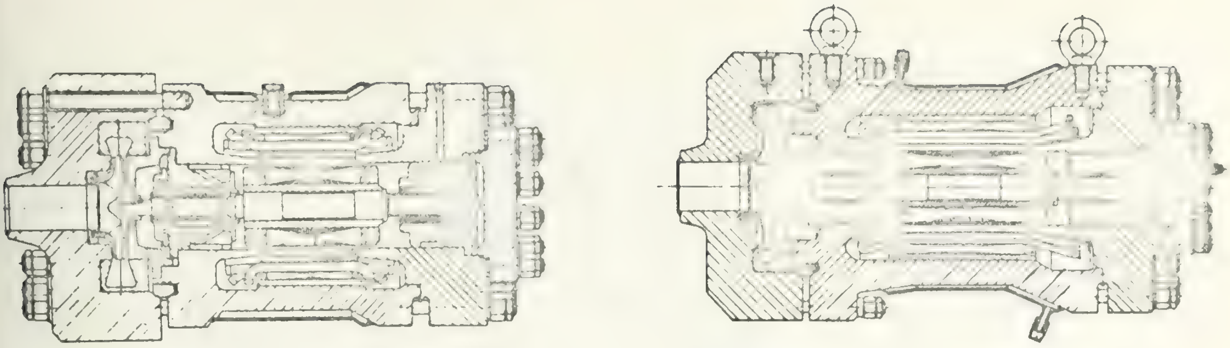
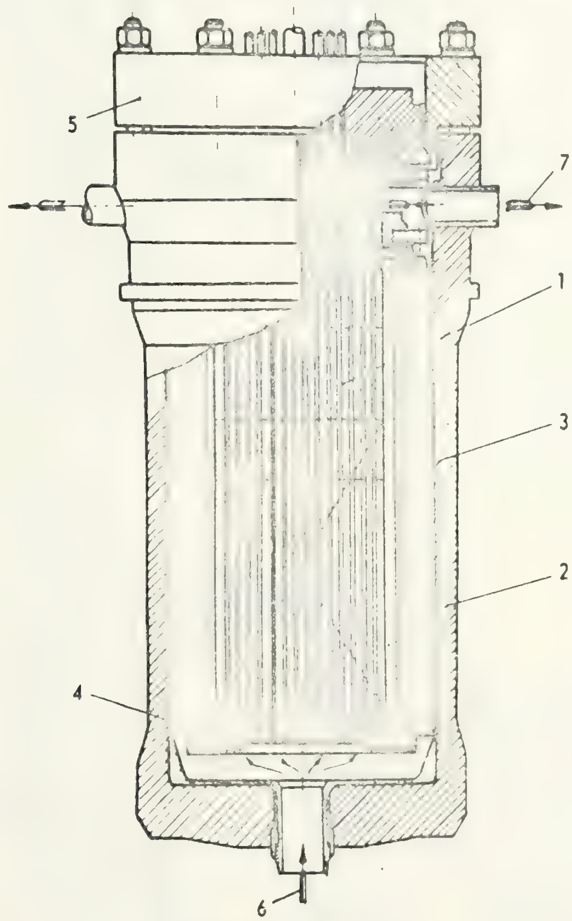
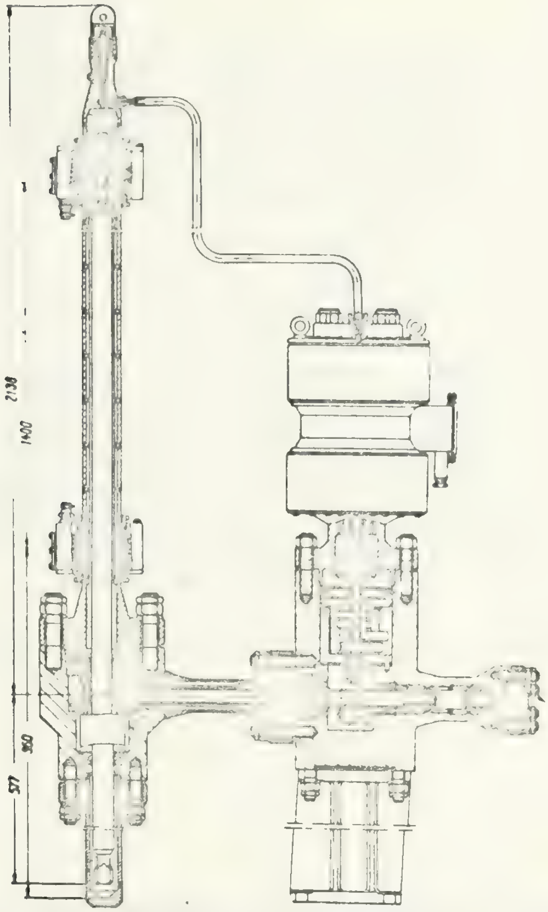


Figure C-3 N.S. LENIN Primary Coolant Pumps

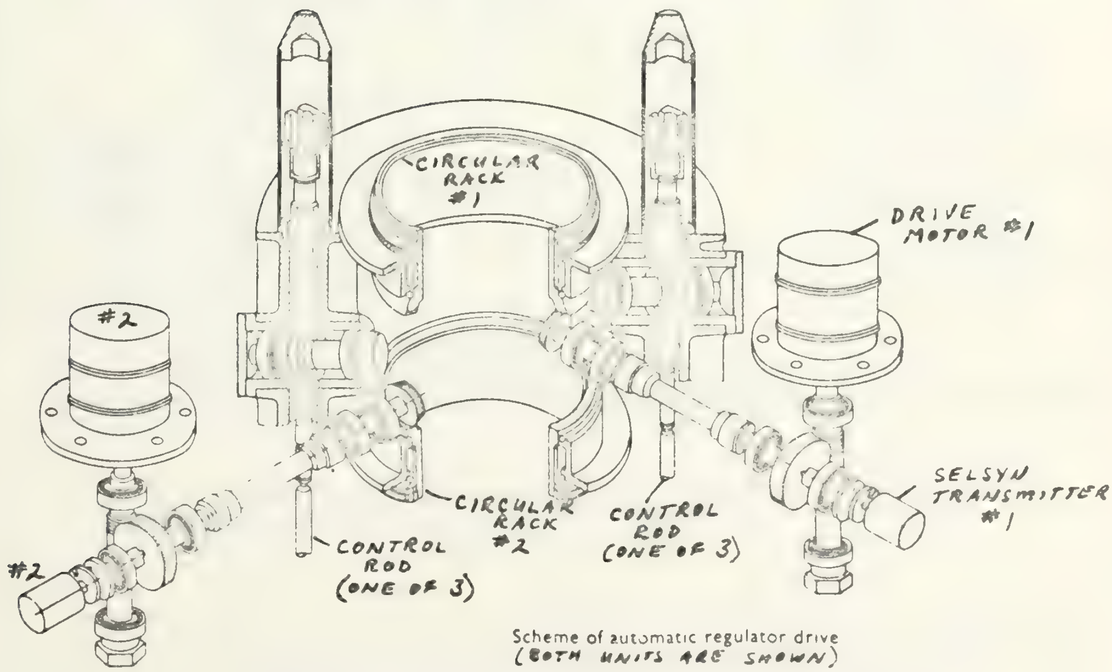


1, Channels; 2, pressure vessel; 3, shielding, 4, lower plate; 5, cover; 6, coolant inlet; 7, coolant outlet

Figure C-4 N.S. LENIN Reactor Vessel and Internals

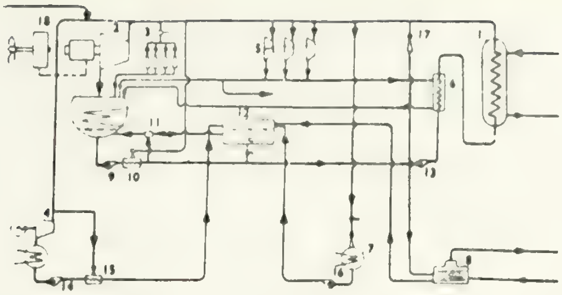


General view of the compensating system



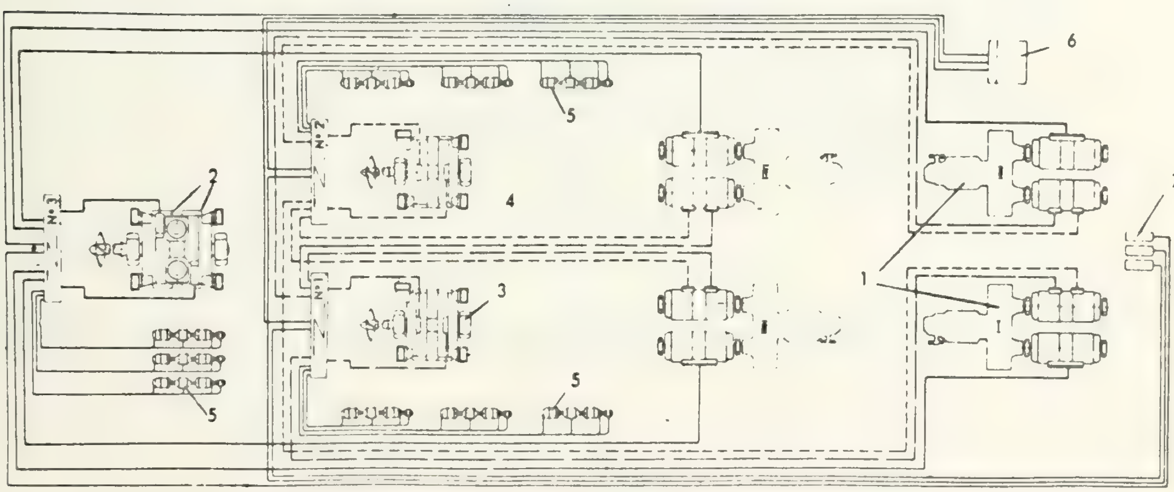
Scheme of automatic regulator drive
(BOTH UNITS ARE SHOWN)

Figure C-5 N.S. LENIN Control Rod Drive Mechanisms



1, Steam generator; 2, main turbogenerator; 3, throttling-wetting device of the steam throttling system; 4, auxiliary turbogenerator; 5, turbine drives of auxiliary mechanisms; 6, feed water heaters; 7, auxiliary condenser; 8, steam generator for general use; 9, feed pump of main turbogenerator; 10, main turbogenerator ejector; 11, level monitor of main condenser; 12, surge tank; 13, feed pump; 14, condensate pump of main turbogenerator; 15, ejector of the auxiliary turbogenerator; 16, condensate pump of the auxiliary condenser; 17, steam cooler; 18, propeller shaft power plant

Figure C-6 N.S. LENIN Propulsion Plant



1, Main turbogenerator units, 2, midship propeller shaft power plant; 3, side propeller shaft power plant No. 1; 4, side propeller shaft power plant No. 2; 5, excitation units; 6, remote control panel; 7, remote control stations

Figure C-7 Propulsion Motor and Turbogenerator Electrical Interconnection Diagram -- N.S. LENIN

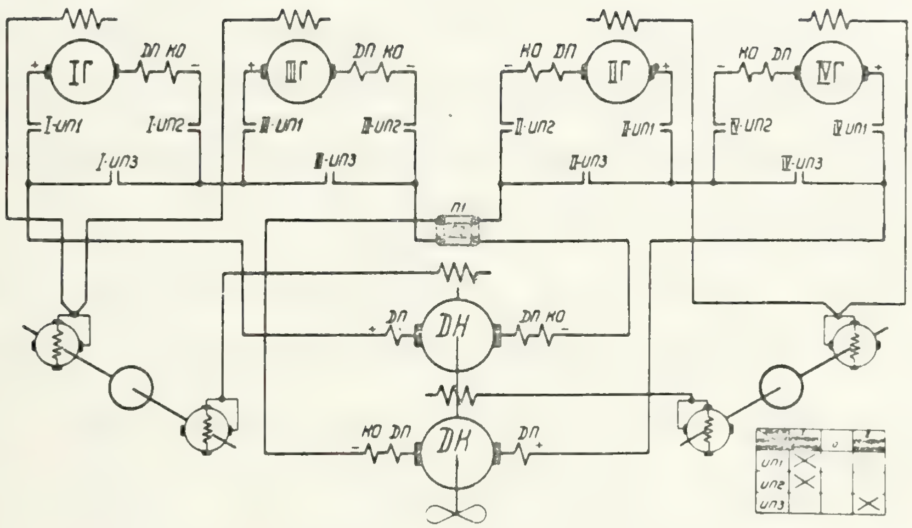


Figure C-8 N.S. LENIN Propulsion Motor Electrical Diagram

D. N.S. MUTSU (ref's. 44 through 54, 60 through 65)

1. GENERAL --

MUTSU is an 8,350 gross ton, 2,400 deadweight ton, 10,000 SHP cargo ship built by Japan for transporting nuclear fuel, for training nuclear plant operators and to gain experience in the design, construction, and operation of shipboard nuclear power plants, thereby enhancing Japan's current position as a leading country in the shipbuilding and maritime industry. The original intent to build an oceanographic survey ship was abandoned, chiefly due to the economic necessity to recover at least a part of the ship's rather large capital investement; the choice of a nuclear fuel transport ship was based on choosing the smallest ship among those capable of bearing a reactor (to minimize the required capital investment) and choosing a ship that would have a "guaranteed" future cargo consistent with national goals (uranium resources are very scarce in Japan, while nuclear reactors are becoming her main source of energy.) Japanese governmental administrative control of the ship's design, construction, and operation is provided by the Japan Nuclear Ship Development Agency, Minato-ku, Tokyo. The only portions of the propulsion plant not produced in Japan are as follows:

Nuclear Fuel (USA)

Control Rod Drive Mechanisms (American Machine and Foundry, USA)

Containment Vessel Penetrations (Scott, West Germany)

Main feed Pumps (Coffin Turbo Pumps, USA)

Moisture Separator (Centrifix)

The reactor plant of MUTSU was designed by Mitsubishi Atomic Industries, Inc.; it is a "loop type," pressurized light water design, similar in basic configuration to that in the N.S. SAVANNAH, except that vertical, U-tube steam generators are used in place of the horizontal tube steam generators installed in SAVANNAH. The 36 MWt reactor will drive the single screw vessel at a service speed of 16.5 knots; design reactor core endurance at this speed is 145,000 sea miles (8,790 effective full power hours).

Construction of MUTSU by Ishikawajima-Harima Heavy Industries Co., Ltd., of Tokyo, began November 27, 1968; the ship was launched June 12, 1969; and the conventional portion of the ship completed in May, 1970. Sailing by auxiliary boiler power to her base harbor and namesake city, Mutsu, located near the northern end of the Japanese mainland, the ship had the nuclear plant installed by Mitsubishi Atomic Power Industries, Inc. Completion of the ship is scheduled for early 1972 after which it will make experimental voyages for 2 years, followed by its intended use for carrying cargo and training reactor plant operators. Additional details of MUTSU are as follows:

Length Over All	426 ft
Beam	62 ft 5 in.
Depth/Full load draft	43 ft 4 in./22 ft 7 in.
Cargo capacity	176,600 ft^3
Displacement, full load	10,400 tons

Complement (including 3 reserve)	22 officers, 37 ratings, plus 20 scientists/ trainees
Compartmentation standard	2 compartment

Reliability of the propulsion plant is enhanced by machinery redundancy, dispersal of vital equipment, and application of the following design criteria (assumed to occur independently) for all propulsion plant equipment:

List and roll (3-9 cycles per min)	60 degrees
Trim and pitch (4-15 cycles per min)	20 degrees
Vertical acceleration	1 $\pm$ 1.0 g
Fore and aft and lateral acceleration	1.0 g

2. SHIP ARRANGEMENT AND STRUCTURE --

The general arrangement of the MUTSU is shown in Figure D-1. The ship is divided into 10 major compartments by 9 transverse bulkheads; any 2 of these compartments can be flooded without loss of the ship, complying with Annex C, "Recommendations for Nuclear Ships," of the 1960 Safety of Life at Sea Convention. Longitudinal bulkheads separate the reactor room and the reactor auxiliary room from the sides of the ship by a distance more than 1/5 the ship's beam on both sides for collision protection of the reactor containment vessel and its contents.

The propulsion plant is arranged in 4 major compartments. The reactor room, enclosed by heavy, concrete bulkheads for radiation attenuation, is located approximately amidships to minimize the vertical accelerations due to pitching. The

reactor auxiliary equipment room, forward of the reactor room, contains some of the reactor auxiliary equipment. The engine room, aft of the reactor room, contains the main turbine and electrical generators. The auxiliary boiler room, aft of the engine room, contains the auxiliary boilers and electrical generators. The main control room, from which reactor and engine room machinery are normally controlled, is located above the engine room on the after part of the upper deck. The main propulsion plant can be remotely controlled from the wheelhouse as well.

The transverse metacentric height of the ship in the intact condition was selected such that, for any 2 compartments flooded, this height will be at least 2 in., thereby ensuring adequate damaged stability for the floating ship. The likelihood of unsymmetric flooding is reduced by cross-flooding ducts connecting the void spaces on both sides of the reactor and reactor auxiliary rooms through the double bottom.

The hull is designed with a longitudinal framing system for the upper deck and most of the double bottom, and a transverse framing system for the rest. Protection of the reactor plant from collision is provided by heavy structure in the voids outboard of the reactor and reactor auxiliary rooms; this structure consists of 6 thick-plated decks, as shown in Figure D-2. Probability of collision is reduced by installation of high performance navigational equipment and a large rudder capable of 45 degrees angle on each side for improved low speed maneuverability.

The reactor plant is protected from grounding damage by a double bottom of increased depth containing a built-up lattice structure designed to absorb grounding energy by buckling. All living quarters are situated forward of the reactor auxiliary room and the cargo holds are located near the bow and the stern, as shown in Figure D-1. The amidships-mounted radar mast serves as exhaust stack for the reactor and propulsion plant ventilation system. The aft stack, located above the auxiliary boiler room, exhausts gases from the auxiliary boiler and diesel electric generators when these units are operating.

3. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

a. Containment Vessel --

The 270 ton reactor containment vessel, shown in Figure D-3, is an airtight vertical cylinder constructed of high tensile steel plates (60-72 kg/cm^2 tensile strength, Nippon Kaiji Kyokai Category II, Class D) and high tensile steel forgings (60-73 kg/cm^2 tensile strength, IHI-IF-80 Standard), with an ID of 32 ft 10 in. and an inside height of 34 ft 7 in. The upper, spherical, bolted-on portion forms a cupola containing the control rod drive mechanisms and provides a clear opening of 16 ft 1 in. for refueling and reactor servicing. Designed to withstand, at a temperature of 374F, an internal pressure of 177.6 psig or an external pressure of 42.6 psig, the vessel is 1.42 in. thick in the cylindrical portion and 2.36 in. thick in the spherical portion. The bottom of the vessel contains 2, 35.5 in. ID, spring-loaded

pressure-equalizing valves which open inward at a pressure difference of 28.4 psid to prevent rupture of the vessel by hydrostatic pressure in the event the ship sinks in deep water.

The weight of the vessel is borne by 2 circular, skirt-shaped foundations mounted on the upper, double bottom plating. Lateral and fore and aft movement of the vessel is prevented by 4 keys mounted on the shoulders of the vessel; these keys mate with keyways mounted on the reactor room bulkheads. All supports are designed to accommodate thermal expansion of the vessel. Arrangement of piping and electrical penetrations through the vessel walls is shown in Figure D-3. Normal access is via a bolted manhole with a 31.5 in. clear opening, located on the forward side of the vessel. Internal vessel temperature is maintained at 113F normally, and not over 140F, by an air conditioning system.

b. Radiation Shielding --

The radiation shield of MUTSU consists of primary and secondary shielding, both of which are shown in Figures D-3 and D-4, plus primary ion exchanger shielding not shown. The primary shielding, designed to prevent significant neutron activation inside the containment vessel and to permit access to the containment vessel 24 hours after shutdown of the reactor, consists of a 39.4 in. thick annular ring of concrete (144-232 lb/ft^3) surrounding the upper portion of the reactor pressure vessel, and a cylindrical primary shield tank containing alternating concentric layers of iron and water surrounding the lower portion of the reactor pressure vessel. The external

surfaces of the primary shield tank and the lower head of the reactor pressure vessel are shielded with lead.

The secondary shield, also shown in Figures D-3 and D-4, is composed of: concrete structures of varying density on the bulkheads surrounding the containment vessel; lead slabs about 7.5 in. thick and polyethylene sheets about 3.94 in. thick on the upper part of the vessel; and a double bottom water tank beneath the concrete structure to attenuate radiation reflected from the bottom of the ship.

In addition to the primary and secondary shielding, the primary ion exchangers, located in the reactor auxiliary equipment room, are shielded with lead about 5.9 in. thick. Moreover, the double bottom tankage can be filled with water to permit drydock work in the vicinity of the reactor room. Shielding weights are as follows:

Primary shielding	250 tons
Secondary shielding	2000 tons
Primary ion exchanger shielding	10 tons
Total shielding weight	2260 tons

The radiation shielding in MUTSU was designed to maintain personnel radiation dose rates at full power in the following ranges (see Figure D-3):

In contaminated spaces (short term entry for machinery inspection and/or repair)	$\leq$ 12 rem/yr
In controlled spaces and monitored spaces	$\leq$ 5 rem/yr
In radiation safety spaces (living quarters, cargo holds, etc)	$\leq$ 0.5 rem/yr

c. Primary System --

The primary system consists of 2 piping loops which circulate light water primary coolant in a closed system, transferring heat generated in the core to the secondary coolant in the steam generators. As shown in Figure D-5, each loop contains one 285 KW, vertically installed, centrifugal, canned motor primary coolant pump, one U-tube, vertical drum steam generator (shown in Figure D-6), and one check valve to prevent bypassing of the core in the event a pump in one loop stops. The primary coolant pumps have auxiliary windings to allow half speed operation for decay heat removal after reactor shutdown.

Primary coolant flow rate through the core is 5000 gpm; core inlet temperature is 518F and outlet temperature is 545F. Each primary coolant pump is rated for 2500 gpm at a pressure head of 50 psig. The check valves have small holes drilled in their discs to allow a small amount of natural circulation flow for emergency decay heat removal. Total steam generator output at full power is 142,400 lbs./hr saturated steam at a pressure of 568 psig. Steam generators and primary coolant pumps are supported by a circular shelf mounted on the bottom head of the containment vessel.

d. Auxiliary Systems --

i) Pressurizing System -- A single, vertical, cylindrical pressurizer, connected to one of the primary loops, maintains primary pressure at 1560 psig by controlling the output of its 130 KW total capacity electric heaters to the

92 ft^3 water volume and by providing in-surge spray to the 92ft^3 steam volume during power transients. A relief valve attached to the top of the pressurizer prevents primary system overpressure.

ii) Volume Control System -- This system performs several functions (see Figure D-5):

-- extracts primary coolant at a low flow rate from the suction side of one primary coolant pump, cools the coolant in a regenerative heat exchanger and 2 non-regenerative after-coolers, reduces its pressure, purifies it in 2 parallel, shielded ion exchangers located in the reactor auxiliary equipment room, and delivers it to the surge tank.

-- stores excess coolant from primary system heatup and spare coolant for primary system makeup in the surge tank.

-- adds hydrogen to the surge tank water to effect recombination of oxygen in the primary coolant, thereby reducing corrosion in the system.

-- delivers coolant at a flow rate controlled by pressurizer level, from the surge tank to the discharge side of the same primary coolant pump, via 1 or 2 charging pumps and the regenerative heat exchanger; part of this water is branched off and used as sealing water for the control rod drive mechanism shaft seals.

-- provides the capability to inject a boron-containing solution into the primary system to keep the reactor shutdown in an emergency situation requiring such action.

iii) Decay Heat Removal System -- provides a means of cooling the primary system without use of primary coolant pumps, by pumping coolant from one loop's hot leg through the 2, volume control system non-regenerative after-coolers to the opposite loop's cold leg.

iv) Emergency Cooling System -- performs 3 functions:

-- to prevent core meltdown due to loss of primary coolant, provides emergency injection of coolant into the reactor pressure vessel from the primary shield tank via the 2 safety injection pumps and from the deaerator and a distilled water tank via the 2 auxiliary feedwater pumps.

-- to minimize pressure buildup in the containment vessel due to a major primary coolant leak, provides steam-quenching spray into the containment vessel from the emergency water storage tank via 2 container spray pumps.

-- to prevent core damage due to loss of normal decay heat removal means, provides feedwater to the steam generators from the emergency water storage tank via the emergency decay heat removal pump to allow heat removal by natural circulation of primary coolant and venting of steam to atmosphere from steam generators.

v) Sampling System -- permits periodic analysis of primary and secondary coolant for chemistry and radioactivity levels.

vi) Component Cooling System -- provides closed cycle fresh water cooling for the following components via 1 of

2 installed pumps; this system is in turn cooled by an open cycle sea water cooling loop with 2 parallel pumps:

primary shield tank

primary coolant pumps

non-regenerative after-coolers

pressurizer relief valve steam-quenching tank

containment vessel normal and emergency air conditioner cooling coils

sampling system heat exchangers (2)

vii) Waste Disposal Systems -- consist of 2 basic systems:

-- The liquid waste disposal system provides a means for collecting and storing radioactive liquids in the 92 ft^3 pressurizer relief valve steam-quenching tank (this also receives the discharge of the safety valves on the steam generators, the charging line, and the volume control letdown lines), a 14 ft^3 drain tank (which collects reactor vessel flange and other primary coolant leakage), a containment vessel sump tank, 2 (350 ft^3 each) medium level ($>10^{-4}$ $\mu c/ml$) waste tanks, and 2 (350 ft^3 each) low level ($<10^{-4}$ $\mu c/ml$) waste tanks. The system is sized to contain all expected radioactive liquids on a 6 month cruise at sea and to pump these liquids for disposal to the shore facility in Mutsu harbor. Solid wastes are also stored onboard the ship for later disposal by this facility.

-- The gaseous waste disposal system provides: 1) ventilation for areas of the ship which might be subjected

to air-borne radioactivity; and 2) off-gas disposal for the radioactive liquid storage tanks. The reactor room and the containment vessel, the reactor auxiliary equipment room and the rooms on the third deck above it are maintained at a negative pressure by a 7,000 ft^3 /min and a 2,800 ft^3/min fan blowing exhaust air through a bank of high efficiency filters and overboard via the hollow main mast. Off-gas from storage tanks is collected in a manifold and piped to the foot of the main mast where it is introduced into the ventilation exhaust and where all gases going up the stack are diluted with fresh air by a 17,600 ft^3/min fan.

e. The Reactor --

i) The Core --

The core comprises essentially a vertical cylinder with an active height of 41 in. and an equivalent diameter of 45.3 in. As shown in Figure D-7, primary coolant enters the reactor pressure vessel through 2 nozzles just below the vessel flange and flows downward through annular passages formed by the thermal shield between the vessel wall and the core barrel, into the inlet plenum at the vessel bottom. Turning upward through coolant guide tubes, the coolant flows through the 32 fuel assemblies in a single pass, mixes in the outlet plenum above the core, and exits via the 2 outlet nozzles.

The core is shown in cross section in Figure D-8. The inner 12 fuel assemblies contain 3.2 w/o enriched fuel; and the other 20 contain 4.4 w/o enriched fuel; the radial zoning

produces a more uniform thermal neutron flux and a flatter distribution of heat generation in the core. Each fuel assembly (see Figure D-9) is arranged in a square matrix with 11 rods on each side (view B-B). Nine of the 121 rods in each assembly contain burnable poison (boron carbide) dispersed in zircaloy. Spacing between rods is maintained by 4, spring-type, stainless steel grid plates distributed equally along the assembly axis; each rod is individually supported by the bottom grid plate. The fuel assembly is protected by side plates, to which the grid plates are spot welded, and by the top and bottom grid plates.

Each fuel rod consists of 53 dish-type pellets of sintered uranium dioxide in a 44.3 in. long, 0.415 in. diameter, and 0.0157 $\pm$ 0.0008 in. wall thickness stainless steel tube capped with plugs on both ends. The fuel elements are designed to keep the maximum fuel temperature below 5072F (where UO_2 melts) at 130% design reactor power, and to have enough space in each fuel rod to accommodate fission gases without excessive internal pressures. With a loading of 2.77 tons of UO_2, the core is designed to have a life of 13,500 MWD, giving an average fuel burnup of 5,500 MWD/tonne. The core is operated in such a way as to maintain constant average primary coolant temperature. It is designed to be refuelled by the batch method.

The 12 control rods, shown in Figure D-8, are cruciform-shaped and are provided with zircaloy followers to prevent thermal flux peaking in the rod channels. These rods and their

followers are guided in the core by cruciform-shaped slots in the upper and lower core supporting plates. The active section of each rod is made of a bundle of small diameter Ag-In-Cd alloy neutron absorber rods; this section is attached to the follower and to the upper structure with tie plates.

ii) The Reactor Pressure Vessel --

This vessel, shown in Figure D-7, is a vertical cylinder of 18 ft 6 in. overall height, 5 ft 9 in. ID, and 3.67 in. wall thickness, with a flanged and bolted hemispherical top and a welded bottom. The vessel is constructed of plates and forgings of manganese molybdenum carbon steel; internal surfaces are clad with a 0.236 in. layer of extra low cobalt stainless steel for corrosion protection. Leakage from the vessel top flange is prevented by 2, O-ring gaskets seated between the flange faces. The vessel is supported by 4 legs anchored to the primary shield tank structure.

The thermal shields (see Figure D-8) around the core protect the reactor vessel wall from radiation-induced thermal stresses and from severe radiation damage. These shields are supported by brackets directly welded to the lower end of the vessel wall. The core barrel and core holddown barrel are hung from the inner circumferential edge of the pressure vessel flange.

f. Nuclear Instrumentation --

Neutron-measuring instrumentation in MUTSU consists of 8 channels covering the entire neutron flux range of the reactor with 3 ranges: startup range, intermediate

range, and power range. Each channel is composed of a neutron detector, an electronic amplifier, and indicating, recording, or integrating instruments. The 8 detectors are installed at mid-core height in the lower part of the primary shield tank, through vertical tubes which penetrate the tank top and the lower of the 2 concrete primary shields.

Two channels, utilizing BF_3 detectors, provide startup range information from 10^{-8} to 10^{-1}% maximum reactor power; a third channel with a BF_3 detector is an installed spare. Two other channels, utilizing compensated ion chambers, cover the intermediate range from the neutron level of the in-core source to maximum reactor power; a third channel with a compensated ion chamber detector is an installed spare. The 2 power channels, with uncompensated ion chamber detectors, provide information up to 150% maximum power. The 2 power range and the 2 intermediate range channels are normally used together for neutron level indication during power operation.

g. Control Rod Drive System --

Each control rod is driven by a rack and pinion device coupled to the single drive motor by reduction gearing. The 12 driving devices and the single drive motor are supported by the stuffing tubes welded to the reactor vessel head and by the platform mounted on the vessel head around these tubes. Lifting the rods compresses heavy springs in the drive devices which provide sufficient force to scram the rods in the required time in any ship position up to 60 degrees from the vertical. Automatic reactor scram is provided for the

following abnormal conditions:

- -- high startup rate in the intermediate range
- -- high neutron flux (125% power)
- -- low primary coolant pressure
- -- high primary coolant temperature
- -- low primary coolant flow rate
- -- high or low pressurizer water level
- -- initiation of emergency cooling injection spray
- -- control rod drop
- -- deviation from the specified control rod pattern
- -- excess angle of inclination
- -- high temperature of control rod drive mechanism seal water
- -- manual initiation of scram

4. POWER PLANT DESCRIPTION; PROPULSION SYSTEM --

a. The Main Propulsion Unit --

The main engine is a cross compound, saturated steam turbine rated at 10,000 SHP; a double reduction gear drives the single, 5-bladed screw at 200 rpm. Because of the constant average temperature reactor control program used, inlet steam conditions for the high pressure turbine vary from 888 psig saturated dry steam under no-load condition to 562 psig saturated steam with a steam quality of 99.75% at full power; main condenser vacuum is 28.4 in. Hg. A drain extraction system, 3 stages of steam bleeding, and a moisture separator between high and low pressure turbines maintain low moisture content of the steam in the engine.

Figure D-5 shows the flow diagram for the propulsion system. Condensate from the main and auxiliary condensers is heated to 320F in 3 stages by the combined feedwater heater, the deaerator, and the high pressure feedwater heater before being returned to the steam generators. Steam generator water level is controlled by a 3 element system with inputs of steam flow, feedwater flow, and water level. A bypass line around the main turbine can dump up to 30% of rated steam flow directly to the main condenser to alleviate severe transient influence on the reactor and on secondary system pressure caused by abrupt and large main propulsion turbine speed changes.

The oil-fired auxiliary boiler provides maneuvering and take-home (at 10 knots) power in the event the reactor plant is inoperable. This unit is a water-tube boiler with 2 drums kept at a pressure of 426 psig by reactor secondary steam heating. The auxiliary boiler can be manually lit off and supplying steam to the main turbine within 15 minutes following reactor shutdown; a sequential program ignition system can be actuated from the control room if auxiliary propulsion steam is desired in a shorter time (5 minutes). This boiler can supply 36,000 lbs/hr steam at 442 psig, saturated, providing 1100 SHP; 96,000 gallons of heavy oil is carried for firing this boiler, giving the ship an oil-fired cruising range of 4,000 miles.

b. Electrical System --

Electric power is normally provided by 1 of 2

running, steam, 60 Hz, 450 volt, ac turbogenerator units rated at 800 KW each. Auxiliary electric power is provided, when the normal sources are not operable, by one diesel-powered, a-c generator rated at 720 KW and located in the auxiliary boiler room. Emergency electric power is supplied by a 240 KW, diesel-powered, ac generator located on the main deck. The emergency generator has sufficient power to start up the emergency propulsion system and to power the emergency decay heat removal system and the safety injection system in all conditions.

Power to the nuclear instrumentation is provided by 2 sets of self-exciting, 30 KW, ac generators driven by 40 KW, dc motors which are in turn powered by a 1,000 AH (10 hour rate) battery. The main switchboard is located aft of the reactor room and the emergency switchboard forward of the reactor room for dispersion of vital equipment. Cables leading to important equipment are either themselves dual or separated and run along each side of the ship to redundant equipment.

5. <u>LAND-BASED NUCLEAR SERVICING FACILITY FOR MUTSU</u> --

To minimize the cost of the ship, and to be prepared for servicing future nuclear ships, a land-based nuclear servicing facility was constructed on a 20 acre site in the city of Mutsu. The facility is equipped with a 575 ft by 26 ft pier, a 75 ton crane, and the buildings and equipment required for refueling and other reactor servicing operations, spent fuel and radioactive waste disposal, and radiation control.

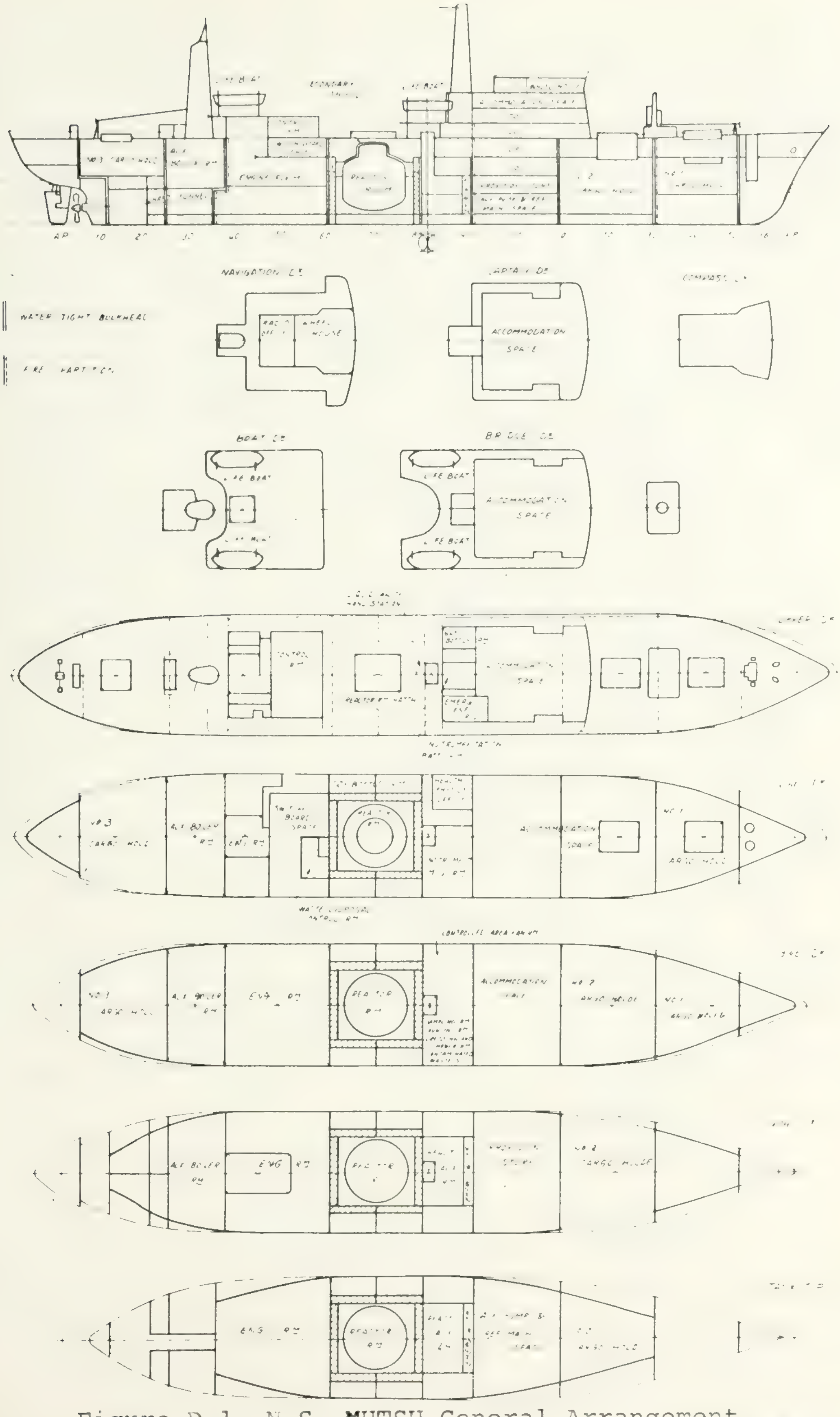


Figure D-1 N.S. MUTSU General Arrangement

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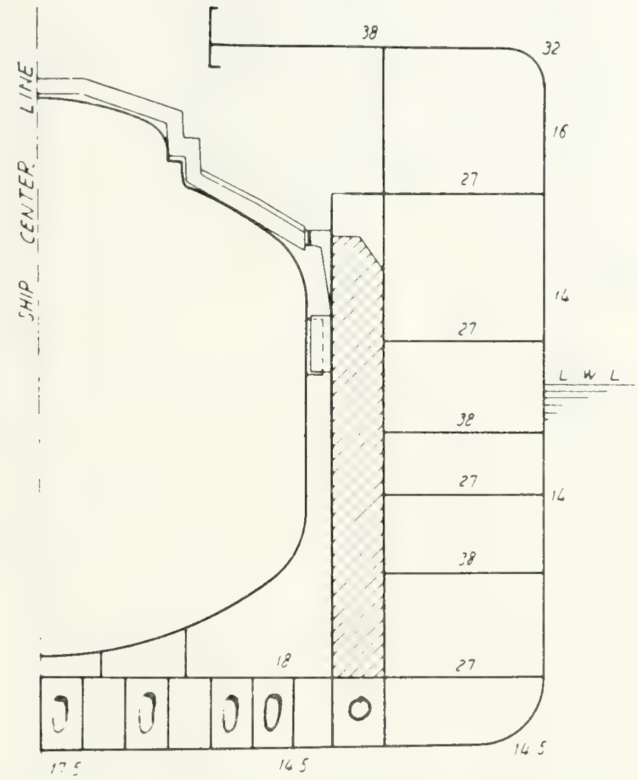

Midship section (numbers show plate thickness in mm).

Figure D-2 N.S. MUTSU Details of Collision Protection

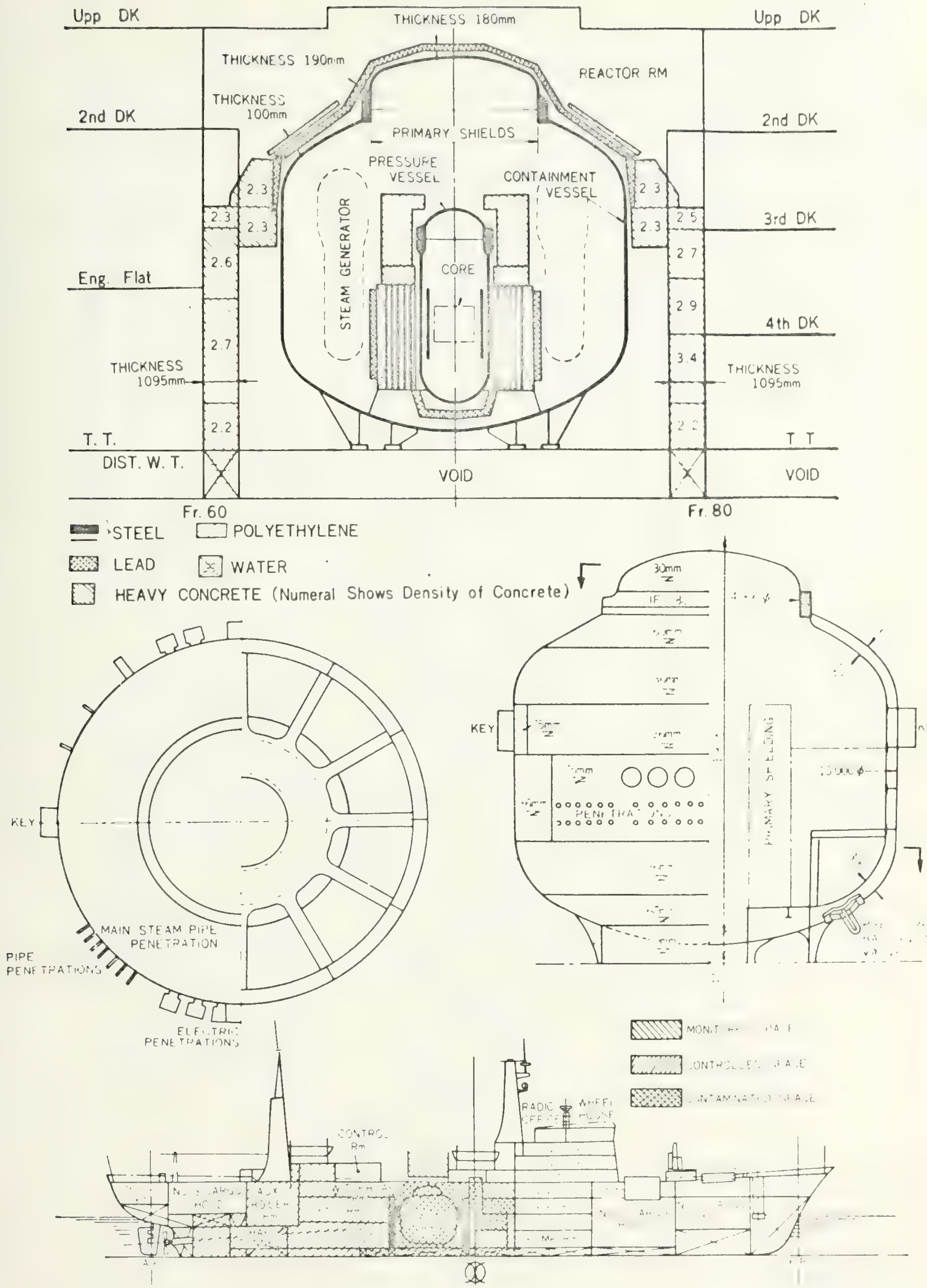

Figure D-3 N.S. MUTSU Reactor Containment Vessel and Radiation Shielding Details

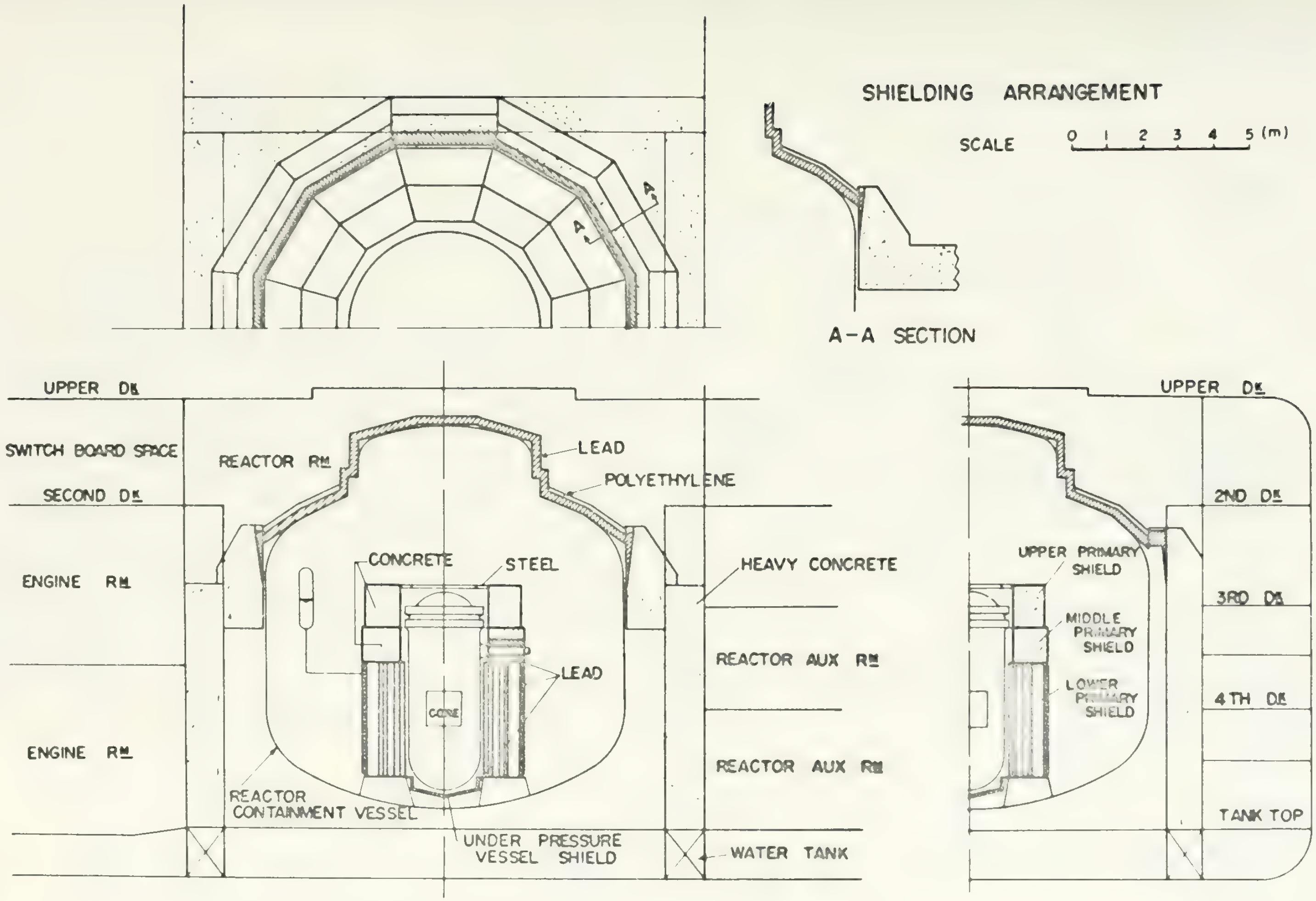

Figure D-4 N.S. MUTSU Radiation Shielding Arrangement

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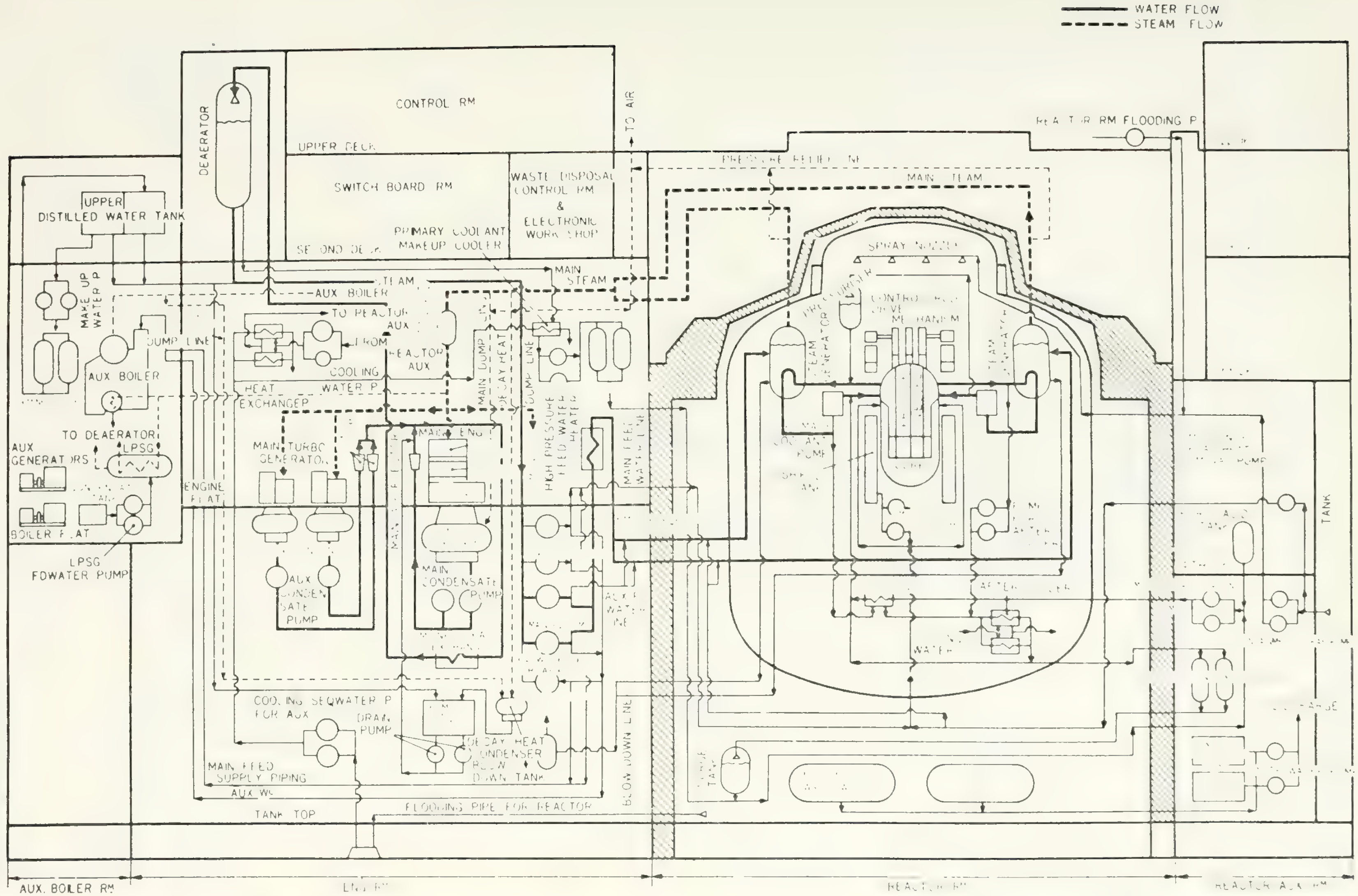

Figure D-5 N.S. MUTSU Propulsion Plant Fluid Systems Schematic Diagram

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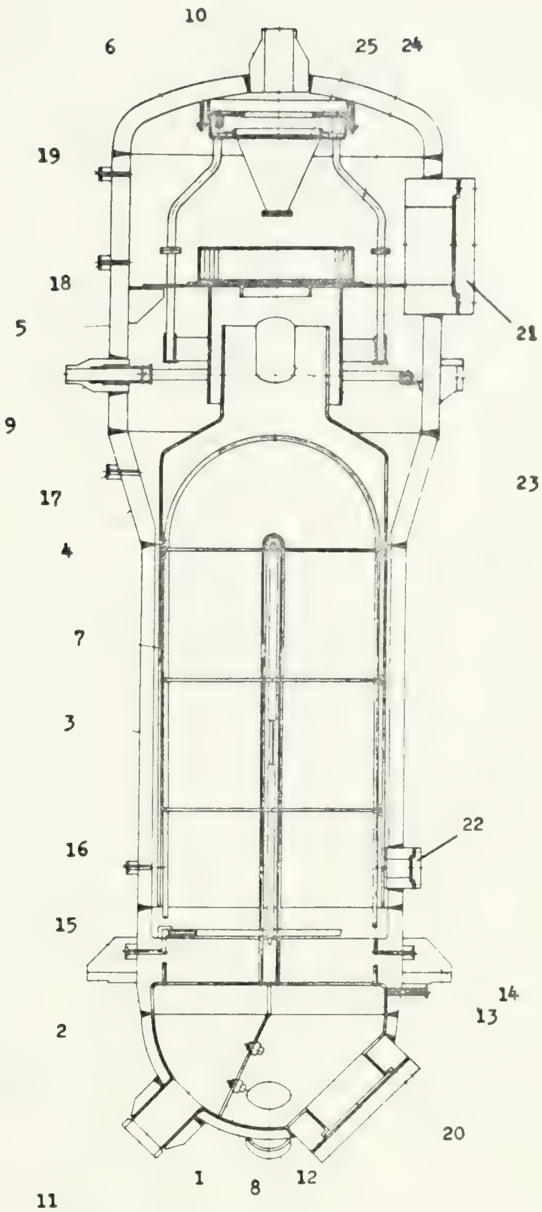
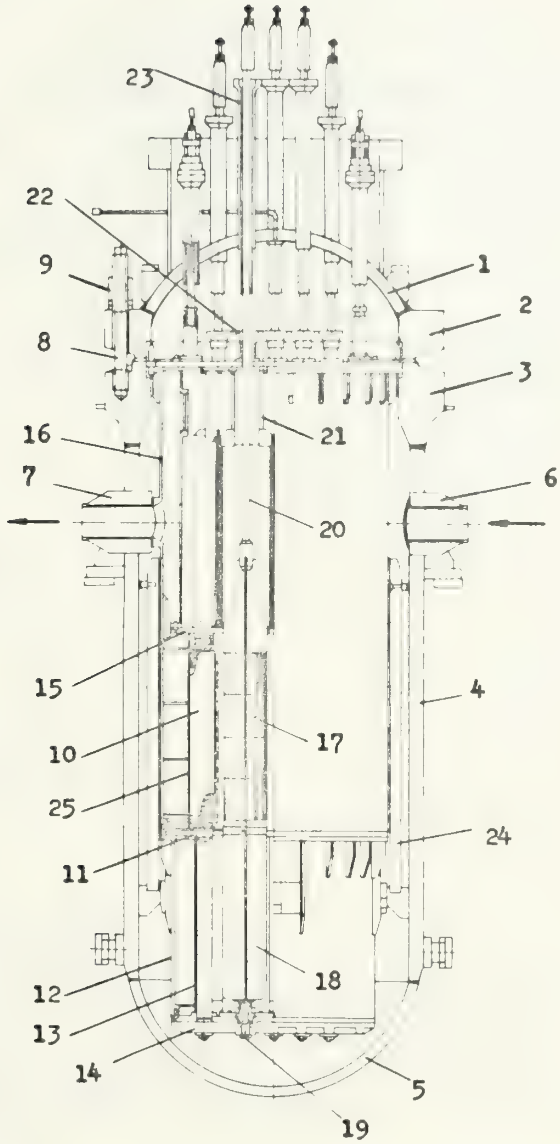


1. Lower head
2. Tube sheet
3. Lower shell barrel
4. Shell cone segment
5. Upper shell barrel
6. Upper head
7. Tube bundle
8. Partition plate
9. Feed water inlet
10. Steam outlet
11. Primary water inlet
12. Primary water outlet
13. Primary water box vent
14. Secondary water drain
15. Secondary water blowdown
16. Water level indicator
17. Water level controller
18. Water level indicator and controller
19. Pressure gauge
20. Primary side manhole
21. Secondary side manhole
22. Hand hole
23. Swirl vane separator
24. Chevron separator
25. Centrifix separator

Figure D-6 N.S. MUTSU Steam Generator



1. Pressure vessel top head
2. Top head flange
3. Shell-flange
4. Pressure vessel shell
5. Pressure vessel bottom head
6. Inlet nozzle
7. Outlet nozzle
8. Stud bolt
9. Washer retaining assembly
10. Fuel assembly
11. Lower core plate
12. Hold down barrel
13. Coolant guide tube
14. Core support casting
15. Upper core plate
16. Core barrel
17. Control rod assembly
18. Control rod follower
19. Dash-pot
20. Control rod drive shaft
21. Control rod guide tube
22. C.R. drive shaft guide tube
23. C.R. shroud
24. Thermal shield
25. Baffle side

Figure D-7 N.S. MUTSU Reactor Pressure Vessel

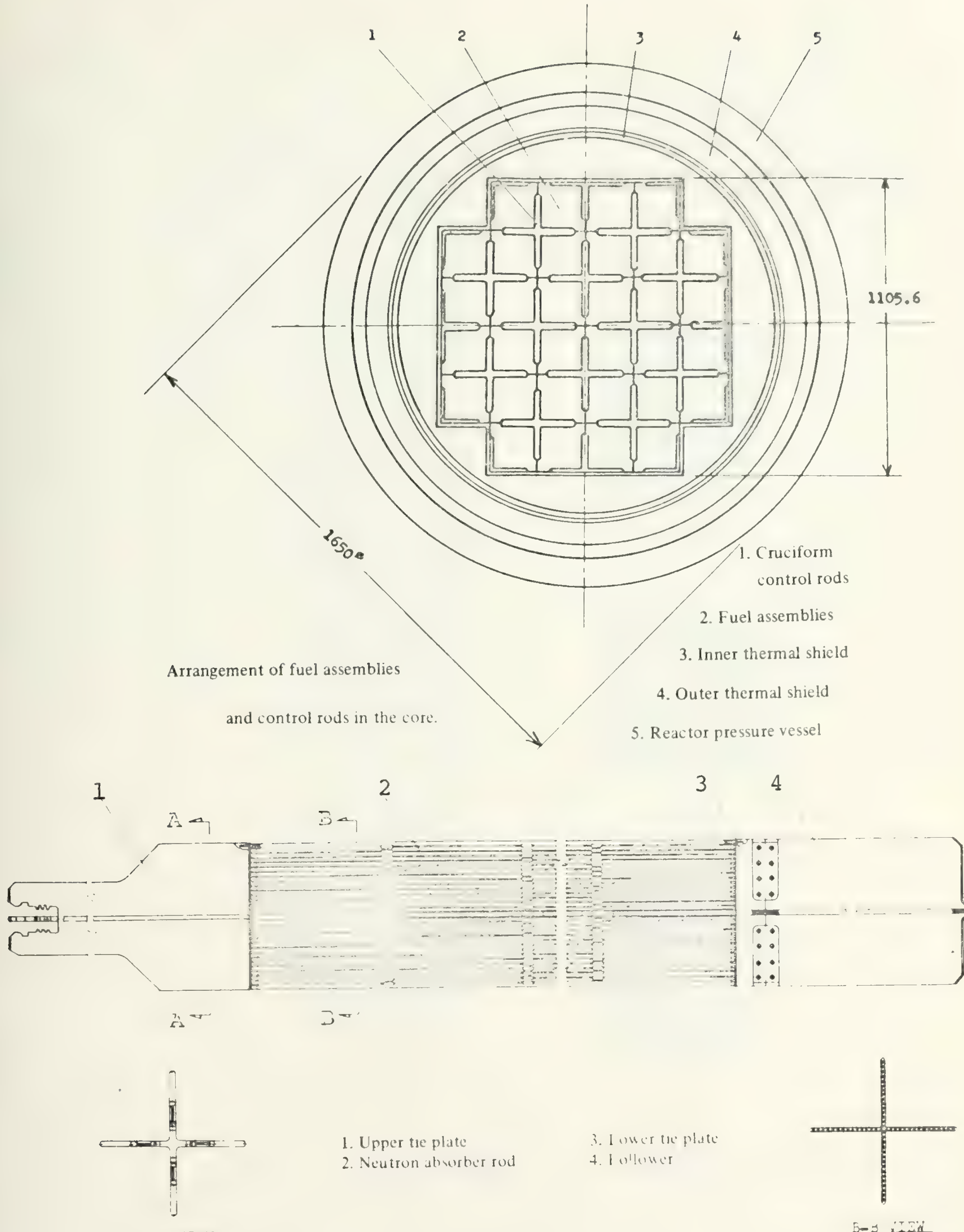
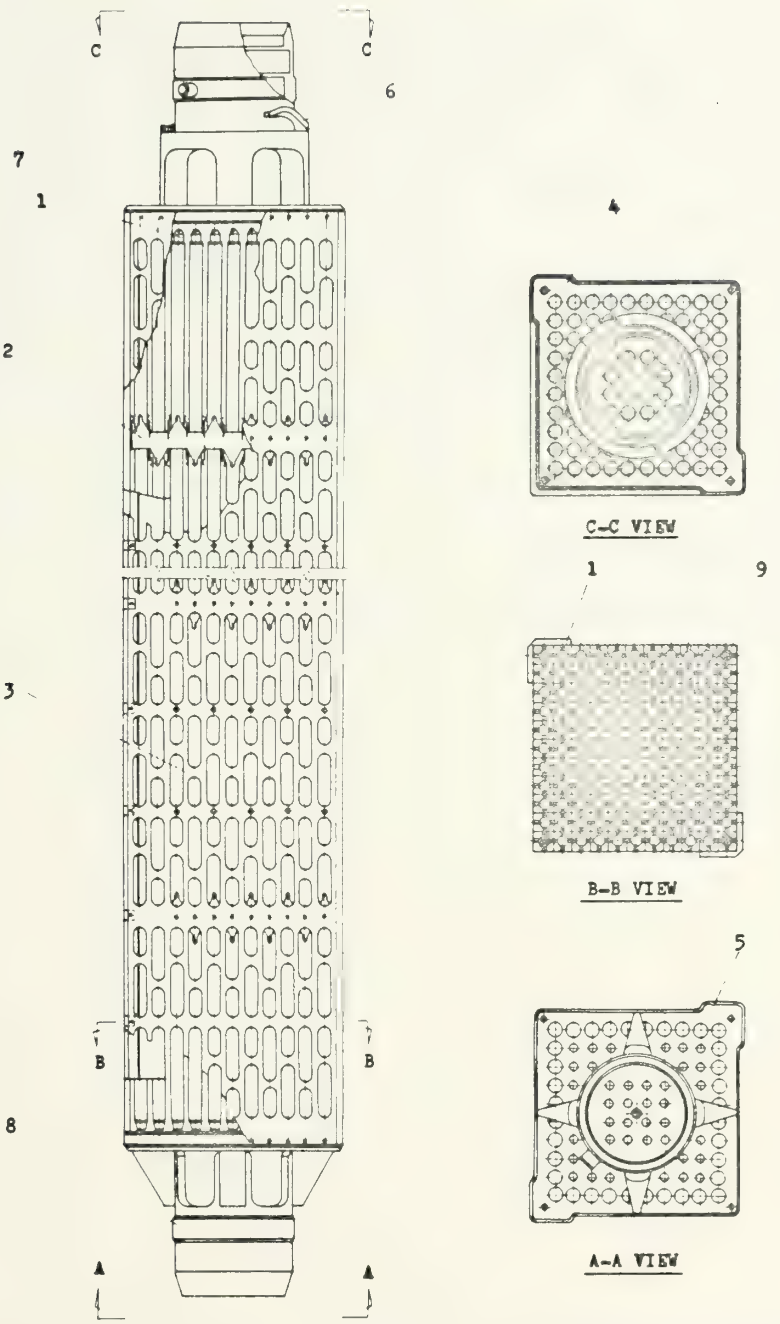


Figure D-8 N.S. MUTSU Reactor Core Cross Section and Control Rod Details



1. Fuel rod 2. Sprin spacer 3. Side plate 4. Upper end plate 5. Lower end plate
6. Coil spring 7. Upper nozzle 8. Lower nozzle 9. Burnable poison

Figure D-9 N.S. MUTSU Reactor Fuel Assembly

E. BABCOCK & WILCOX'S CONSOLIDATED NUCLEAR STEAM GENERATOR (CNSG) -- (Ref's 18, 68, 69, 70)

Babcock and Wilcox is currently completing a major updating and improvement of the CNSG design, based on recent experience gained from construction and operation of the N.S. OTTO HAHN. Since current United States export regulations do not permit inclusion of such technical information in a publication such as this, current details of the CNSG design have been provided to Professors A. Douglas Carmichael of the Ocean Engineering Department and Manson Benedict of the Nuclear Engineering Department. These details may be obtained from either of them for use subject to the restrictions stated thereon.

ANALYSIS OF PAST, PRESENT AND FUTURE APPLICATIONS OF NUCLEAR POWER FOR PROPULSION OF MARINE VEHICLES-COMBINED XIII/XXII THESIS BY JAMES R. BAUMAN, 1972

APPENDIX I. DETAILS OF SPECIFIC NUCLEAR MARINE PLANTS

SECTION E. BABCOCK & WILCOX'S CONSOLIDATED NUCLEAR STEAM GENERATOR (CNSG)

Since current United States export regulations do not permit inclusion of technical information regarding the current CNSG design in this thesis, the details included herein are provided for the use of selected personnel, subject to the restrictions stated below.

> The information contained herein is deemed to be subject to the export licensing provisions of 15 CFR Part 385, the current United States export regulations. Any exportation, as defined in said regulations, of any of this information without consent of the Babcock & Wilcox Co. is expressly prohibited.

E. BABCOCK AND WILCOX'S CONSOLIDATED NUCLEAR STEAM GENERATOR (CNSG) (ref's. 18, 68, 69, 70)

1. GENERAL --

Experience in building the N.S. SAVANNAH emphasized the significant value of fabricating at least the entire primary system in the shops, thereby eliminating from the construction yard the costly and time consuming operation associated with the assembly, welding and flushing of a high pressure stainless steel system. Elimination of this shipyard fabrication operation would greatly reduce the length of time the ship's hull would remain on the building ways, would decrease the ship construction cost, and would increase the willingness of shipyard management to undertake nuclear shipbuilding work by reducing the time necessary for the yard to tie up its facilities with a nuclear ship.

To minimize shipyard fabrication time and cost, and to develop a shipboard nuclear propulsion plant which would consume less of a ship's total volume and weight than was consumed by the propulsion plant of the N.S. SAVANNAH, Babcock and Wilcox has developed the CNSG propulsion plant, with General Electric designing the non-nuclear portion. The CNSG design has progressed through a series of updatings and improvements from the original 20,000 SHP version to the current 120,000 SHP version. The SHP value of 120,000 was chosen on the basis of economic comparisons with fossil-fueled steam propulsion plants and is compatible with that used in the latest high speed containerships constructed;

in these B & W comparisons, 110,000SHP was determined to be the "break-even" power level at which nuclear and conventional propulsion plants would cost the same over the life of the ship. Only the most recent design will be described below; this design reflects recent experience gained through construction of the CNSG-based plant in the N.S. OTTO HAHN.

Basically, the CNSG design substitutes flow passages within the reactor vessel for the bulky, primary system piping, and places the steam generators and primary coolant pumps (but not the motors) within the reactor vessel; primary coolant flow is confined to within the reactor vessel. The absence of large, primary coolant piping that can rupture enhances plant safety and integrity. The final phase of this U.S. Maritime Administration funded design involves preparation of: 1) a firm cost and detailed design, with the high speed containership as the reference hull, and 2) a preliminary safety analysis report to be submitted to the AEC as the basis for a construction license independent of hull type; this phase is scheduled for completion in December, 1972.

2. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

a. Containment Vessel --

Shown in Figure E-1, this vessel is a 1.25 in. thick, carbon steel cylinder 34 ft in diameter and 48 ft high, with ellipsoidal top and bottom, capable of withstanding 90 psig internal pressure. The operating floor, roughly at vessel mid-height, provides lateral support for the centerline-

mounted reactor pressure vessel and supports the pressurizer which is on the centerline and forward. This floor divides the containment vessel into a dry well chamber above and a wet well chamber, or vapor-suppression pool, below. Connecting the 2 chambers are 4, 15-in. diameter vent pipes which direct flashing primary coolant from a pressurizer surge line rupture (the maximum credible accident for this plant) into the vapor-suppression pool. Access is via the 19 ft ID removable re-fuelling head and a double barrier personnel hatch.

b. Radiation Shielding --

Biological shielding (see Figure E-1) is provided by the vapor-suppression pool water, a layer of lead around the reactor vessel, and a 2 ft thick wall of heavy aggregate concrete around the containment vessel.

c. Primary System --

As shown in Figure E-1, the light water primary coolant is circulated within the reactor vessel at a flow rate of 8,080 gpm by 4, controlled leakage or canned motor, axial vane primary coolant pumps whose electrical motors are mounted external to the reactor vessel. Primary coolant at 572.3F enters the core at the bottom, flows upward at an average velocity of 11.6 ft/sec, and enters the pumps at 604.5F where it is forced downward over the 4 parallel sets of annular steam generator tubes guided by 4 sets of cans enclosing all but the tops and bottoms of the steam generator tubes to prevent flow bypassing, and back to the core inlet.

Total steam generator heat transfer area is 5,360 ft^2, made up of 1,200 tubes of 3/4 in. OD, 0.077 in. wall thickness. Feedwater enters each of the 4 steam generator modules at 400F through a tubesheet, flows downward through a 17 ft 6 in. downcomer shielded from direct primary coolant to prevent boiling-induced flow instability, flows upward making 19 horizontal passes through the winding, 94 ft length of each tube and leaves through a tubesheet as slightly (50F) superheated steam at 700 psia, 553F; full power steam flow is 1,224,000 lb/hr at a heat transfer rate of 1,068 M Btu/hr.

d. Auxiliary System --

i) Pressurizing System -- maintains primary system pressure at 1850 $\pm$ 25 psia by controlling electrical input to the 336 KW total capacity heaters in the lower part, and spray flow to the upper part, of the 70 in. ID, 14 ft 2 in. high, 3.19 in. wall thickness, cylindrical pressurizer vessel with ellipsoidal top and bottom; a safety valve set at 2075psia relieves steam pressure to prevent overpressurizing the 2100 psia design pressure vessel.

ii) Makeup and Purification System -- purifies primary coolant at a rate of 30 gpm maximum in 1 of 2 parallel ion exchangers; flow path is from the reactor vessel through 1 of 2 parallel letdown coolers (120F outlet), a set of throttling orifices (150 psia outlet), a fine mesh filter, the ion exchangers and another filter, to a 1,500 ft^3 makeup tank; then from this tank via 1 of 2 multi-stage centrifugal makeup pumps and a throttling valve which controls flow rate

to maintain pressurizer water level. This system can store excess coolant from primary system heatup and return the coolant later for primary system cooldown. Hydrazine and lithium hydroxide are added to the makeup tank to maintain desired primary system chemistry.

iii) Component Cooling System -- transfers heat to sea water from the following components: letdown coolers; vapor-suppression pool coolers; primary coolant pump motor coolers; control rod drive mechanism coolers; and decay heat removal coolers.

iv) Decay Heat Removal System -- valved into the primary system after steam generators have cooled it to below 300F (68 psia sat), this system removes core decay heat to 140F within 20 hours by taking coolant from the reactor vessel, through purification ion exchangers and filters if desired, through 1 of 2 parallel pump and decay heat removal cooler circuits and returning it to the reactor vessel; flow rate is 500 gpm. This system can also take water from the vapor-suppression pool and/or the containment vessel sump and inject it directly into the reactor vessel at pressures below 200 psia to prevent fuel rod cladding damage for an extended period following a loss-of-coolant accident.

v) Intermediate-Pressure Injection System -- injects water from the vapor-suppression pool at 100 gpm with 1 of 2 pumps into the reactor vessel plenum beneath the core at pressures below 1200 psia to help prevent cladding failure during a loss-of-coolant accident.

vi) Reactor Containment Cooling System -- maintains containment vessel air temperature below 100F with 1 of 2 coolers cooled by sea water.

vii) Emergency Boric Acid Injection System -- keeps the reactor subcritical in a cold shutdown condition by injecting boric acid into the primary system.

viii) Containment Vessel Spray System -- provides 1,000 gpm spray into the containment vessel dry well chamber from the vapor-suppression pool via 1 of 2 parallel pump and cooler circuits to help minimize the pressure rise during a loss-of-coolant accident.

ix) Ventilation System -- provides a means of replenishing reactor compartment air and of maintaining the containment vessel pressure below atmospheric for loss-of-coolant accident mitigation by reducing the blanketing effect of non-condensible gases in the vessel.

x) Waste Disposal System -- provides onboard tankage, pumping and piping system for collection and storage of liquid wastes and for dilution and discharge of gaseous wastes; solid wastes are manually drummed for disposal by commercial disposal facilities at designated ports of call.

e. The Reactor --

i) The Core --

The design of the 312.5 MWt core is essentially that used for a typical central station pressurized water reactor (PWR) with additional lateral support for fuel rods, to withstand ship motion and machinery vibration,

provided at 2 horizontal planes by spring-type spacer grids and core clamp mechanisms previously developed for PWR cores subjected to seismic disturbances.

The core consists of 45 fuel assemblies in an 8.6 in. square pitch, 7 by 7 array with the corner assemblies missing; equivalent diameter is 64.7 in. As shown in Figure E-2, each 930 lb, canless fuel assembly has an active height of 85.75 in. and consists of an 0.568 in. square pitch, 15 by 15 array of 0.430 in. OD, 0.0265 in. wall thickness zircaloy-4 tubes, 192 of which contain 0.370 in. OD uranium oxide pellets; the unfuelled tubes either contain lumped boron carbide burnable poison (16 tubes) or act as guide tubes (16 tubes) for the 304-stainless steel clad, Ag-In-Cd cluster-type control rods. The function of the central tube can be fuel, poison, or instrumentation depending on location in the core. Lateral support for the tubes is provided by 6, spring-type, Inconel-718 spacer grids.

The 5 year core lifetime is provided by 11,405 kg of uranium, with enrichments as shown in Figure E-2 and averaged at 4.707%, a load factor of 70% full power, and an average burnup at the batchtype refuelling of 35,000 MWd/ton. Movable control rod worth is 0.26Δk/k for the hot core and 0.21Δk/k for the cold core. The 720 lumped burnable poison rods initially control 0.144Δk/k of the total 0.228Δk/k required for the 30,670 equivalent full power hours core life.

Total core heat transfer surface area is 6,950 ft^2 with an average/maximum heat flux of 153,416/410,567 Btu/hr

ft^2 ; design power peaking values used were 1.65 for radial x local, 1.60 for axial, and 2.64 for total. Thermal and hydraulic design criteria are as follows:

1) No central fuel melting at 115% power.

2) A 99% confidence that at least 99.5% of the fuel rods will not experience departure from nucleate boiling (DNB) under continuous 115% power.

3) 100% confidence that at least 99.96% of the fuel rods will not experience DNB under continuous 100% power.

4) Generation of net steam in the hottest channels is permissible, provided steam voids are below the threshold for flow instabilities.

ii) The Reactor Pressure Vessel --

This vessel and its principal dimensions are shown in Figure E-3. The flat head simplifies the bolted control rod drive-head connection and allows removal of the head for refueling or steam generator module replacement without disturbing the pumps. The feedwater and steam nozzles have built-in thermal barriers to increase fatigue life, and are designed (as is the feed and steam piping up to the first shutoff valves) to hold primary pressure in the event the belleville spring-type seal between the tubesheets and the nozzles leaks. Bolted closures at the ends of the nozzles can be removed to permit plugging defective tubes without removing the steam generator module.

f. Instrumentation and Control --

Four identical sets of each of the instrument

assemblies described below are provided:

i) Startup and intermediate range nuclear instrumentation covering the range 10^{-10} to 1.5 full power utilize both counting and Campbelling techniques with a fission chamber to achieve both wide range and high gamma discrimination. The chamber is located in the bottom of a dry well assembly passing between 2 steam generators and terminating just outside and near the bottom of the core barrel.

ii) Power range nuclear instrumentation covering the power range to 1.5 full power utilizes 3, prompt-responding, all-solid radiation detectors bundled and inserted in a wet well that passes vertically through the pressure vessel head and between the core shroud and the core barrel. Each of the 3 detectors is recalibrated once every second with an on-line core thermal power calculation using the same computer (a Varian Data Machine 620/i) that is used for the plant protection system described below.

iii) Thermocouples for measuring primary coolant core inlet (2) and outlet (2) temperatures with 3 installed spares are inserted in the power range instrumentation wet well. A high accuracy Bourdon tube/capacitance gauge instrument measures primary system pressure.

In addition to these instruments, the following control systems are provided:

i) Plant Control System -- maintains constant average primary coolant temperature by moving control rods as necessary during plant operation above 15% power, sensing

steam flow, reactor power, and deviation from desired average coolant temperature.

ii) Plant Protection System -- comprises 3 independent channels, each of which monitors the plant's status and casts a trip/no-trip vote reguarding the unsafe/safe posture of the plant; 2 out of 3 trip votes actuates the appropriate annunciator alarm and light, and protective action if necessary. Input signals from various instruments are monitored every 0.01 to 1.0 sec and appropriate comparisons and calculations made by the digital computer to determine the safety of the particular portion of the plant involved.

3. POWER PLANT DESCRIPTION; PROPULSION SYSTEM --

a. The Main Propulsion Unit --

The description below is for a 2-shaft plant, although 3 or 4 shafts could also be used. Each of the 2, double cylinder turbines accepts 700 psia, 50F superheated steam at full power and delivers 60,000 SHP at 90 to 135 rpm through multiple-drive double-reduction gears to its shaft. In the crossover line between the high pressure and the low pressure turbines the steam pressure is about 55 psia; a static, swirl vane moisture separator removes 80 to 95% of the entrained moisture to improve cycle efficiency and reduce the potential for erosion in the low pressure turbine. Moisture is also removed in both turbines by grooves and dams in stages below the steam dew point.

00290k

Pressure in the sea water scoop-cooled main condensers is 2 in. Hg; feedwater leaves the condenser and is pumped through a Zeolite water conditioning demineralizer. The feedwater is then heated in the gland exhaust condenser where it condenses the steam used to seal the ends of the turbines; it then passes through the first low pressure feed heater where it is heated by steam extracted from the low pressure turbine. A second low pressure feed heater raises the feedwater temperature to 220F, after which it is sprayed into the deaerating feed tank; the vented gases and some steam are led back to the gland seal condenser where the steam is condensed, heat recovered, and the gases vented. The demineralized and degassed feedwater is then pumped back to the steam generators via high pressure feed heaters, where its temperature is raised to 400F by 1 of the 3, 2,650 gpm, 1,000 psid, steam-driven main feed pumps.

Turbine ahead and astern control valves are positioned by a cam shaft driven by hydraulic cylinders governed by electrical or air signals from the bridge-mounted control units.

b. Electrical System --

Electrical power is provided by 2, 2,500 KW ship's service turbo generators, with 2 full-sized 2,500 KW diesel standby generators and a 750 KW emergency diesel generator provided. The steam turbines for the ship's service turbo generators are single casing machines geared to the generators and designed with internal provisions for

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moisture removal similar to those on the main propulsion turbines. Each turbine has its own condenser to allow generation of electrical power in the event main condensers are not operable.

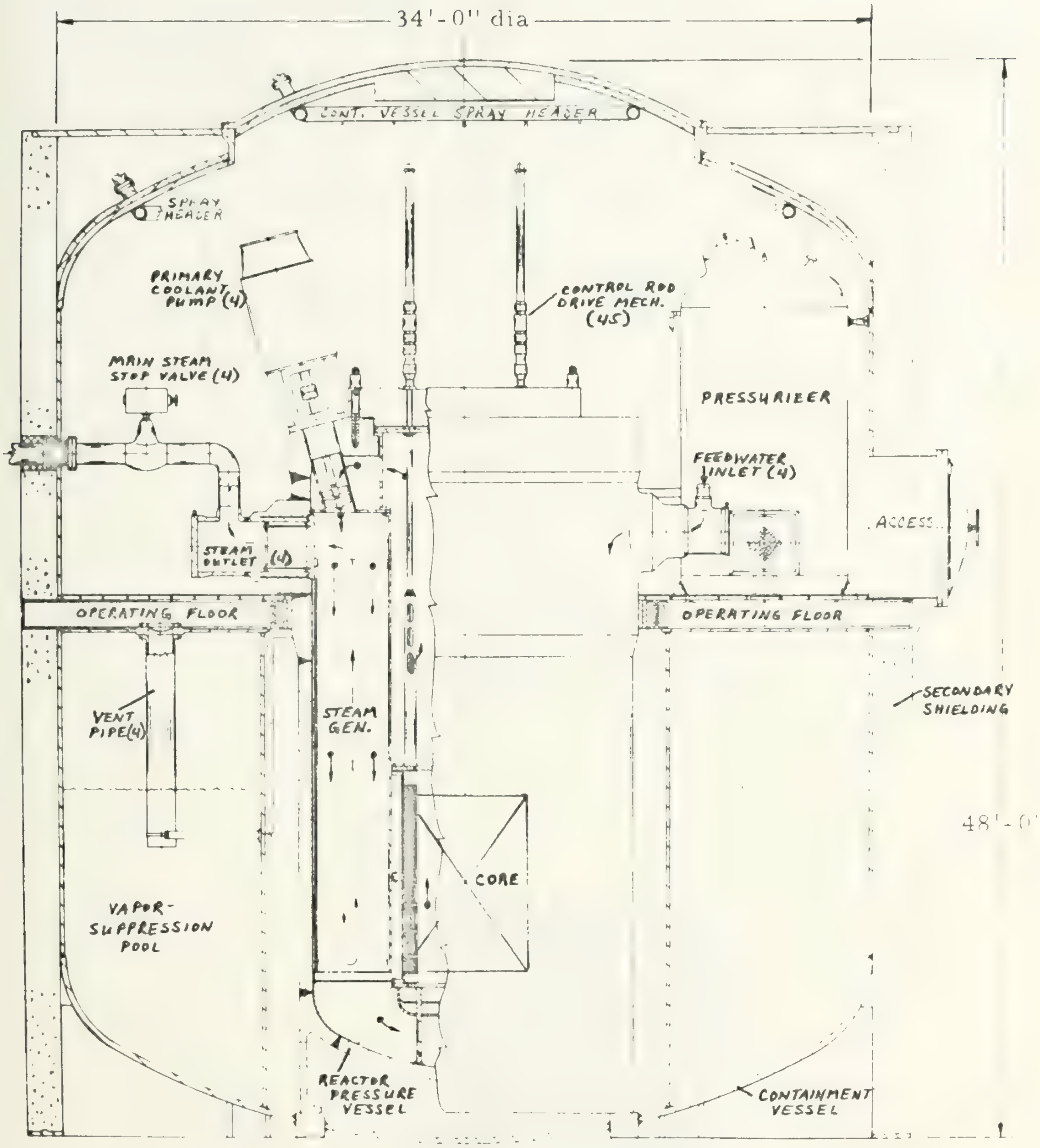

Figure E-1 CNSG Containment Vessel and Internal Equipment

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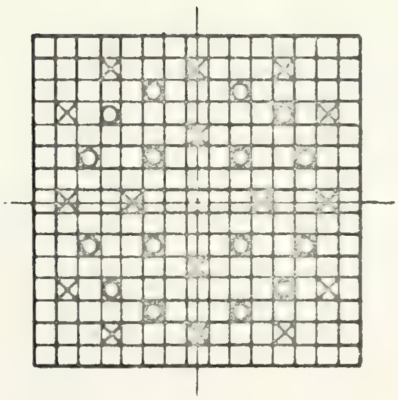
	4.9	4.9	4.6	4.9	4.9	
4.9	4.6	4.6	4.6	4.6	4.6	4.9
4.9	4.6	4.6	4.6	4.6	4.6	4.9
4.6	4.6	4.6	4.6	4.6	4.6	4.6
4.9	4.6	4.6	4.6	4.6	4.6	4.9
4.9	4.6	4.6	4.6	4.6	4.6	4.9
	4.9	4.9	4.6	4.9	4.9	

Enrichment, wt % ^{235}U

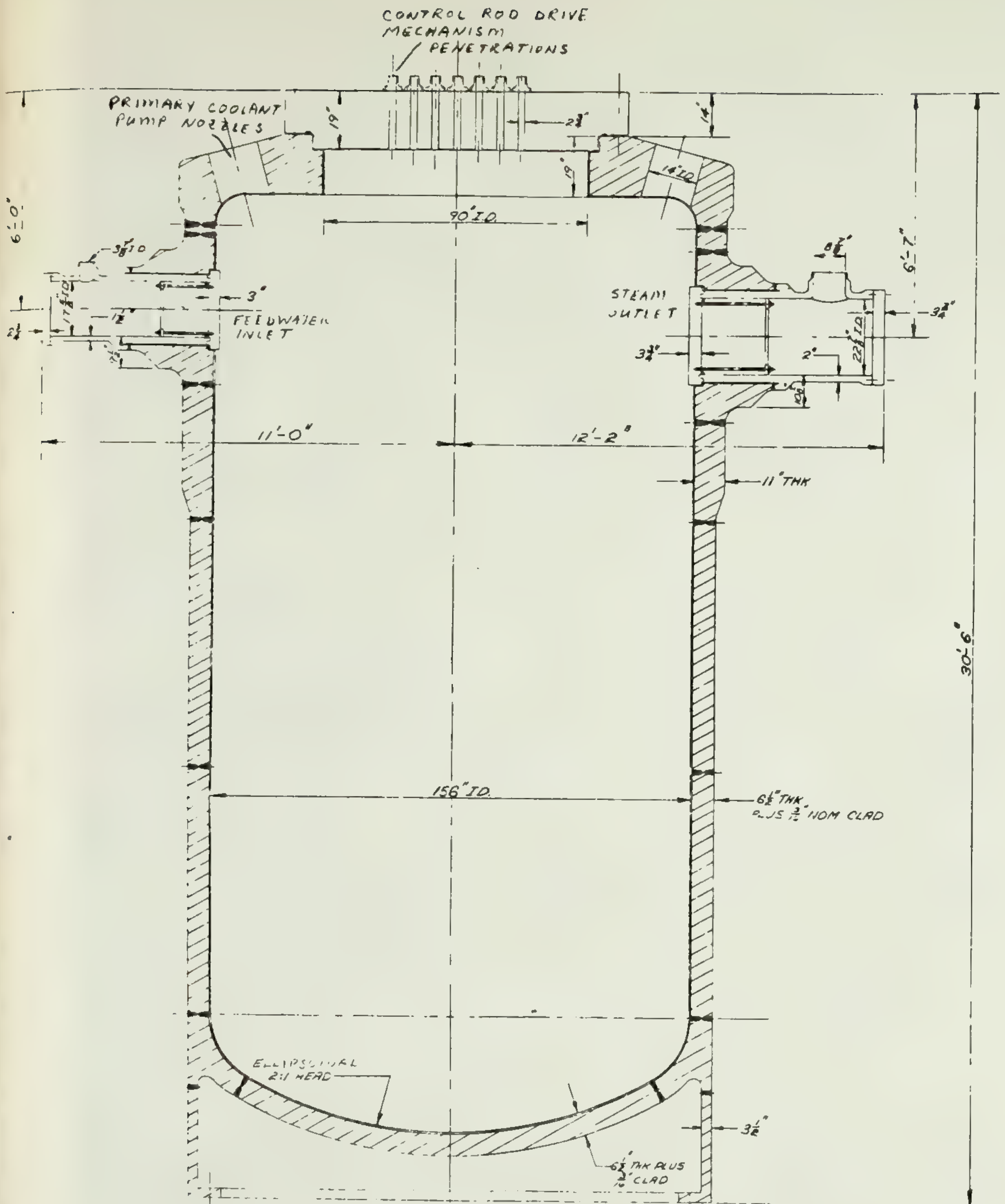


□ FUEL ROD

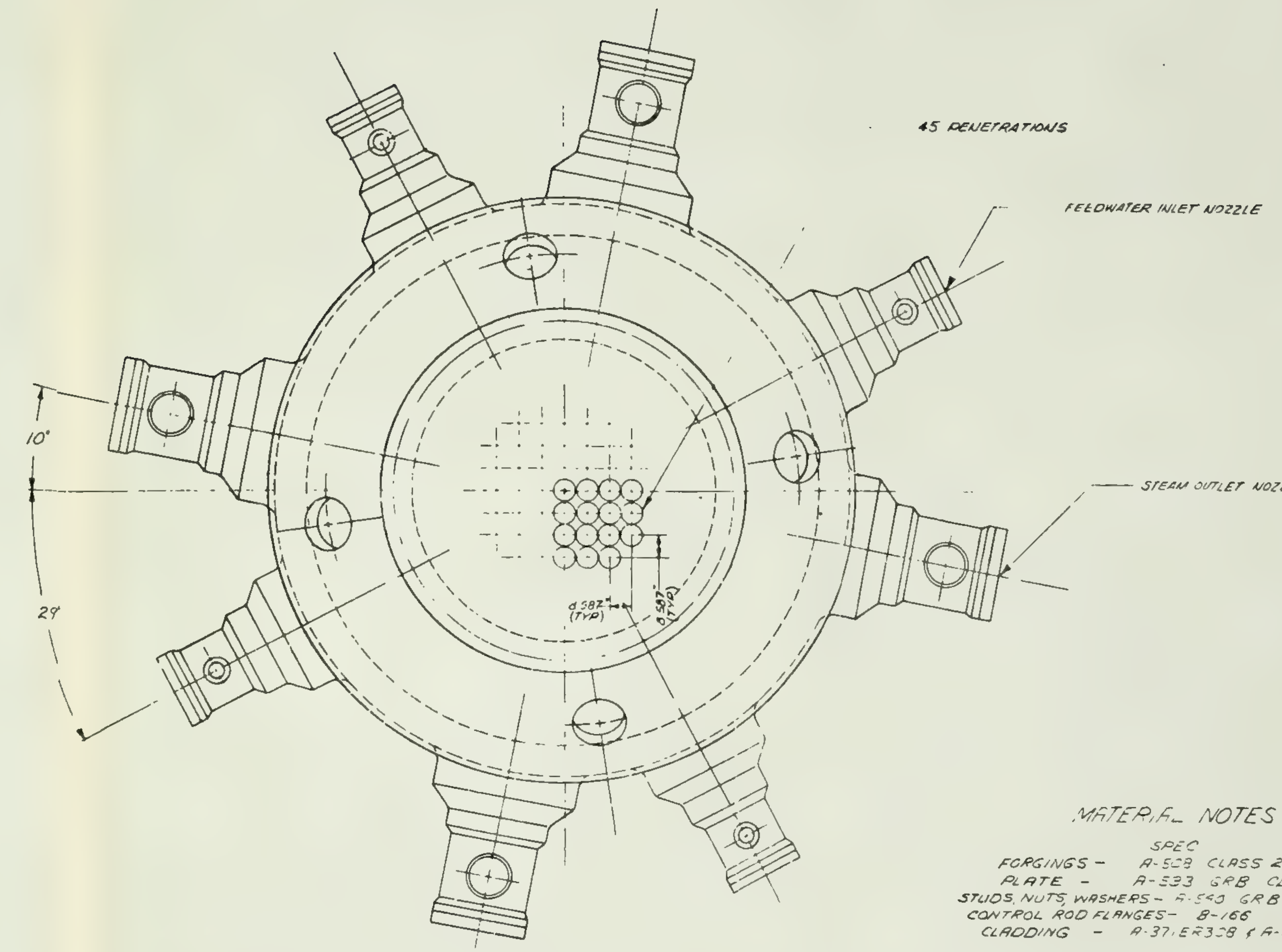
▣ CONTROL ROD

▩ LPB ROD

Figure E-2 CNSG Core and Fuel Element Cross Section



ELEVATION SECTION



PLAN VIEW

MATERIAL NOTES

	SPEC
FORGINGS –	A-508 CLASS 2
PLATE –	A-533 GRB CLASS 1
STUDS, NUTS, WASHERS –	A-540 GRB 24 CLASS 3
CONTROL ROD FLANGES –	B-166
CLADDING –	A-371 ER308 & A-298-E 308

Figure [illegible]-3 CNSG Reactor Vessel

F. GENERAL ELECTRIC'S 630A MARITIME NUCLEAR STEAM GENERATOR -- (ref's. 18, 70, 72 through 76)

1. GENERAL --

Application of the technology developed for the Aircraft Nuclear Propulsion Program between 1951 and 1961 to the design of a nuclear plant for merchant ship propulsion was undertaken in September, 1961 by General Electric under contract to the U.S. Atomic Energy Commission. The resulting plant, the 630A, was designed to produce steam for driving normal steam turbomachinery so as to: 1) require minimum modification of existing installations and technology if used for back fit, 2) utilize existing turbomachinery technology without having to develop closed cycle turbomachinery technology, and 3) achieve greater operating economy. The 630A is about the same size and somewhat heavier than the conventional boiler it can replace.

Since its inception, the design has undergone 5 major evolutions; the first 3 involved an air-cooled, water-moderated core, and the last 2 a helium-cooled, water-moderated core. All 5 design evolutions emphasized features suited to low cost quantity manufacture and pretesting in a factory, and rail shipment in a small number of major subassemblies for easy and rapid modular installation aboard ship; factory-made cable runs and the use of electrical pin-connectors further facilitate installation. Reduction of the length of time the ship's hull remains on the building ways by providing for rapid modular installation of the reactor plant can

greatly reduce the cost of nuclear ship construction and increase the willingness of a shipyard to tie up its facilities with a nuclear ship.

The 630 Mark V design is described below. Although this description is that of a 27,300 SHP plant, any power level between 2,000 SHP and 120,000 SHP may be realized utilizing this base design. The 630A is basically a high temperature, gas-cooled (helium at 830 psia), water-moderated reactor plant which can produce steam at any desired level of pressure and temperature up to 1500 psig and 1000F (Pressurized water reactors require very high primary coolant pressures to produce steam at even 600F). Overall thermal efficiency of the plant using the highest steam conditions is 33.7%; weight of the containment vessel and its contents and external shielding is about 460 tons.

2. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

a. Containment Vessel --

The containment vessel, shown in Figure F-1, is a low-carbon steel (SA 201B (A300)), cylindrical vessel, 39 ft 1 in. high and 12 ft ID, fitted with 2 horizontally oriented, 6 ft 1 1/2 in. long cylinders with hemispherical heads to enclose the helium circulators. The vessel is designed to withstand an internal pressure of 535 psig at a temperature of 650F; this capability is sufficient to contain the helium coolant in the vessel's 1,400 cu ft net volume, in the event the reactor pressure vessel should

rupture. Containment vessel internal temperature is maintained at a level suitable for instruments and control systems by an air-to-water heat exchanger. The vessel also contains fire detection and fire fighting systems. Access to the vessel is provided by a manway in the ellipsoidal bottom head and by the refueling/service plug in the top head.

b. Radiation Shielding --

As shown in Figure F-1, the radiation shielding consists of: 1) concentric cylinders of steel, lead, and water at atmospheric pressure containing 0.6 w/o boron, and 2) the top shielding plug, containing steel, lead, and moderator water. This shielding is designed to maintain radiation dosage below 5 rem per year for radiation personnel working in adjacent compartments and below 0.5 rem per year for other personnel on the ship. In addition to this shielding, the upper portion of the containment vessel is shielded with a layer of lead to reduce external dose rates to acceptable levels in the event the reactor pressure vessel should rupture after a major fuel cladding failure.

c. Primary System --

All primary flow takes place within the reactor pressure vessel; helium was chosen for the primary coolant for 4 qualities important to mobile nuclear systems: 1) high specific heat, so low pumping power, 2) high thermal conductivity, so smaller heat transfer surface area, 3) noncombustible and not chemically reactive, so safe to handle, and

4) low induced radioactivity, so lower shielding weight.

Helium at 828 psia, 553F enters the core at the top, flows downward through the core primary flow channels at a flow rate of 3900 lb/min, and enters the superheat section of the steam generator at 824.2 psia, 1200F. Leaving the steam generator at 822.7 psia, 550F, the helium passes through the two 1.009 pressure difference coolant circulators and exits at 830 psia, 553F to return to the top of the core via an annular flow channel between the core and the reactor pressure vessel. Total primary pressure drop is 7.3 psia; total primary gas loop volume is 490 cu ft. A maximum gas temperature of 1200F was selected to assure required fuel element life and to reduce design problems with structural and boiler tube materials by maintaining at a low value the amount of material creep that had to be considered.

The 2 gas circulators are each driven by a direct-coupled, 250 HP, ac electric motor supported on gas bearings; circulator speed is controlled by varying the frequency of the power supply. The gas circulators are completely contained within the containment vessel to eliminate the need for shaft seals. In an emergency, the plant can produce 18,000 SHP utilizing air pressurized to 830 psia as the primary coolant; in such a situation, primary temperatures would be approximately the same as for helium, while flow rate would increase to 14,400 lb/min and pressure difference across the circulators to 1.012.

The steam generator is a once-through water-tube unit consisting of 252 tubes, each 162 ft long, 0.625 in. OD and 0.085 in. wall thickness, formed into a serpentine configuration in the vertical plane. Three tubes form a clip; each clip is formed into an involute curve in the radial plane so that the tube spacing normal to the airflow direction is equal at all radial locations. Two separate feedwater and steam collection headers are provided so that a failed tube can be isolated by securing header isolation valves while maintaining 50% propulsion power. All tube ends are accessible from the bottom of the assembly for ease of plugging failed tubes. Radiation levels in the bottom of the containment vessel and direct radiation damage to the tubes are reduced by a chevron-type shield assembly between the core and the steam generator. Full power steam flow is 172,800 lbs/hr; steam drum steam conditions at this power are 1535 psia, 1005F, 405F superheat, with a feedwater temperature of 415F and a condenser pressure of 1.5 in. Hg.

d. Reactor-Shield-Plug Assembly --

The reactor core complex, the top shield plug, and the control rods (see Figure F-1) are assembled, checked out, and handled as a single unit called the reactor-shield-plug assembly. The reactor-shield-plug assembly is enclosed within and forms the top head of the reactor pressure vessel (the reactor pressure vessel is SA 212B mild steel, designed for 900 psig internal pressure at a temperature of 700F).

Refueling is accomplished by removal of the reactor-shield-plug assembly into a shielded container and installation of a replacement assembly; ship downtime required for refueling has been estimated, based on experience with the Aircraft Nuclear Propulsion Program test reactor at the National Reactor Testing Station, to be about 5 days. The spent assembly is returned to a service plant for refueling, refurbishment, checkout, and eventual shipment for reinstallation in a 630A plant. Each of the units making up this assembly is described below; this description is in 2 sections, corresponding to the 2 different design reactor-shield-plug assembly developed for use with the 630A plant.

i) The Calandria-Type Reactor-Shield-Plug Assembly--

The calandria-type reactor vessel, shown in Figures F-1 and F-2, is a 4.50 in. thick incoloy cylindrical tank 80.0 in. OD and 66.5 in. high, closed at each end by a tube sheet. The vessel is suspended inside the reactor pressure vessel from the shield plug by 12 moderator exit tubes (5.0 in. OD, 0.183 in. wall thickness, rolled and seal welded to the tube sheets) and the 180 control rod guide tubes (0.700 in. ID, 0.04 in. wall thickness) and is designed to withstand an external pressure of 900 psig.

The 60.5 MWt core consists of 109 fuel cartridges arranged in a hexagonal pattern with a total UO_2 inventory of 7,050 lbs and a total U-235 inventory of 312 lbs; active height and equivalent diameter are 42.00 in. and 48.25 in. respectively. The center 61 fuel cartridges are spaced on a 4.4 in.

pitch and the outer 48 cartridges on a 4.625 in. pitch, thereby varying fuel-to-moderator ratio radially to flatten the radial power distribution; each cartridge contains 55 pin-type fuel elements the size and spacing of which are shown in Figure F-2d (the smaller OD pins are used on the periphery to reduce overall cartridge diameter).

Each fuel pin contains a 6 in. plenum at the top (see Figure F-2c) for fission product gas accumulation; the remainder of the pin is filled with a 10 g/cc matrix of 5.0w/o U-235 enriched UO_2. A 0.020 in. wall thickness stainless steel cylinder around the fuel pin bundle provides the outer surface for the helium flow channel and maintains a stagnant, insulating helium gap 0.040 in. wide between the fuel and the moderator tube. The fuel pins are suspended from the support plate at the top of the cartridge; the bottom support plate positions the pins and provides clearance for axial and radial thermal expansion. The calculated core lifetime is 15,000 MWD/MT, or 17,350 effective full power hours. Total core heat transfer area is 1,860 sq ft and average heat flux is 102,000 Btu/hr sq ft.

To reduce neutron leakage, the core is surrounded top and bottom by 10 in. thick water reflectors and radially by a 4 in. thick annular beryllium reflector (see Figure F-2b); the beryllium reflector outer surface is cooled by a 0.25 in. wide moderator water flow channel between the reflector and the 0.5 in. thick, 56 in. high zircaloy flow divider. Two borated steel thermal shields are provided to reduce reactor

vessel and containment vessel thermal stresses due to gamma heating and to reduce metallurgical embrittlement due to neutron irradiation; a 2 in. thick shield is inside the reactor vessel and a 2.5 in. thick shield is in the helium flow path between the reactor vessel and the reactor pressure vessel.

The SA 212B mild steel top shield plug (see Figure F-1), in addition to providing support for the control rod drive mechanisms, and shielding for the area above the core, and acting as the reactor vessel top head, serves as inlet and outlet plenum for the 250 psia light water moderator. The moderator, cooled to 235F in the moderator heat exchanger by steam generator feedwater, is pumped by 1 of 2 moderator pumps at a flow rate of 2,200 gpm to the upper section of the top shield plug which acts as the moderator inlet plenum and provides a housing wherein groups of control rods are connected together to be driven by a single drive mechanism. From this plenum the moderator flows downward through the core via the 135 shim control rod drive guide tubes, reverses direction and flows upward through the core via the interstitial flow channels between fuel cartridges (outside the moderator tubes), and flows via the 12 moderator exit tubes from the reactor vessel to the lower shield plug which acts as the moderator outlet plenum. From this plenum, which contains a 7 in. thick lead gamma-attenuation shield, the moderator returns at 255F to the moderator heat exchanger, completing the flow circuit.

All shield plug internal surfaces exposed to water are clad with incoloy to minimize corrosion. A tube extending to the reactor vessel bottom through 1 of the moderator exit tubes ensures the capability of removing all moderator water from the vessel if desired.

Three groups of control rods are provided: shim (135 rods), dynamic (9 rods), and safety (36 rods). Shim rods are used to compensate for reactivity changes over core life, while dynamic rods are used for reactor control; safety rods are fully withdrawn against spring pressure during plant operation and are used for scram protection only. All rods consist of a matrix of vibration-compacted B_4C powder (chosen for its low cost and low helium release rate), with a minimum density of 1.70 g/cc, in an incoloy tube; dimensions are shown in Figure F-2d.

Shim and dynamic rods are driven by electric motors; safety rods are withdrawn pneumatically against spring force and are latched, fully withdrawn, by electric solenoids. Since flooding of the core with water (seawater or moderator) reduces core reactivity by increasing resonance capture in U-238 (more than needed to compensate for the reduction in neutron leakage from 10% to 2-3% caused by the presence of the water), no separate control rods are required for core flooding; safety rods alone will maintain the core subcritical in a flooding situation. Loss of coolant flow, e.g. due to gas circulator failure, results in maximum fuel cladding temperatures below 2200F due to the heat transfer to the

relatively cool moderator; these temperatures are well below the melting temperature of 2550F.

ii) The Tube-Type Reactor-Shield-Plug Assembly --

By eliminating the calandria and simplifying the shield plug design, the tube-type assembly (shown in Figure F-3) results in lower fabrication costs, while not changing the method of refueling or the operating conditions and system specifications described above. As shown in Figure F-4, the moderator in this design is contained in the center of each fuel cell, rather than in the interstitial region between fuel cells, by a tube extending from the shield plug inlet plenum down into the core; this tube and the hexagonal-shaped, longitudinal support can discussed below form concentric flow channels in which the moderator flows. This arrangement provides a somewhat better heat release path, so lower peak temperatures, for the loss of coolant flow accident. A gas-cooled reflector and 2 thermal shields surround the core radially and are supported by a structural shell which is in turn supported by the shield plug.

The core contains 151 fuel cartridges arranged in a hexagonal array on a 3.98 in. pitch. Active core height is 42.00 in. and equivalent core diameter is 51.3 in. Each fuel cartridge contains a central control rod and 1 or more zircaloy shim tubes used to displace moderator water to effect flattening of the core radial power distribution. Each fuel cartridge contains 38, 0.390 in. OD, with 0.015 in. thick incoloy cladding, pin-type fuel elements filled with a 5 w/o U-235

enriched UO_2 fuel matrix except for a 6 in. plenum at the top for fission gas accumulation. Arranged in a hexagonal array on a 0.460 in. pitch, the pins are suspended from the top support plate mounted on the moderator tube; the bottom support plate positions the pins and provides clearance for radial and axial thermal expansion. A thin, hexagonal-shaped can provides longitudinal support for the fuel cartridge and serves as the inner boundary of the helium flow passage.

e. Plant Instrumentation and Control --

The 630A plant is controlled by 3 systems:

i) The Reactor Startup System -- monitors reactor power between source level and 10% full power, and displays power level and rate of change of power to the operator.

ii) The Power Range System -- automatically positions control rods to maintain constant core outlet gas temperature between 10% and 100% full power; regulates feedwater flow rate to maintain desired steam conditions of temperature, pressure, and flow.

iii) Safety System -- monitors various plant parameters and provides warning indications and/or safety action as necessary to ensure plant safety.

Reactor power is measured by 7 nuclear sensors located in 7 of the 12 moderator water exit tubes. Core exit helium temperatures are measured by thermocouples located in the instrumentation harness directly below the reactor.

3. POWER PLANT DESCRIPTION; PROPULSION SYSTEM --

Since the references available to the author do not provide a propulsion plant heat balance for the 630A Mark V plant, a heat balance for the 630A Mark IV plant is presented as Figure F-5 for information. Major differences between the Mark IV and Mark V designs include:

1. Primary system --

   a. 1,266,722 vice 234,000 lb/hr helium flow rate

   b. 374 vice 822.7 psia helium steam generator outlet pressure

   c. 67.4 vice 60.5 MWt core heat output

2. Secondary (steam) system --

   a. 193,878 vice 172,000 lb/hr steam flow rate

   b. 955 vice 1005F steam temperature from steam generator

   c. 880 vice 1535 psia steam pressure from steam generator

   d. 407.2 vice 415F feedwater temperature to steam generator

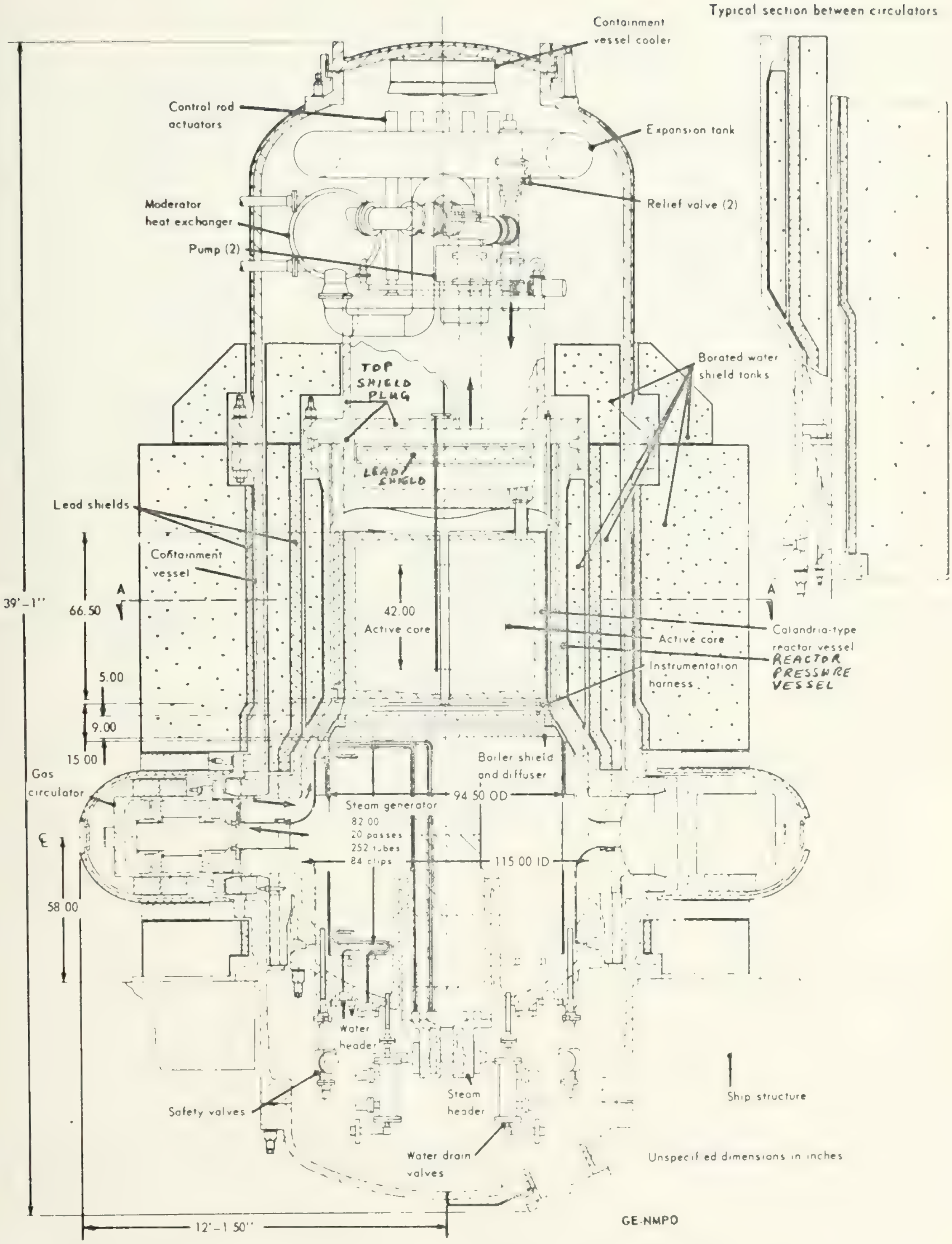


Figure F-1a 630A Mark V Nuclear Steam Generator Cross Section

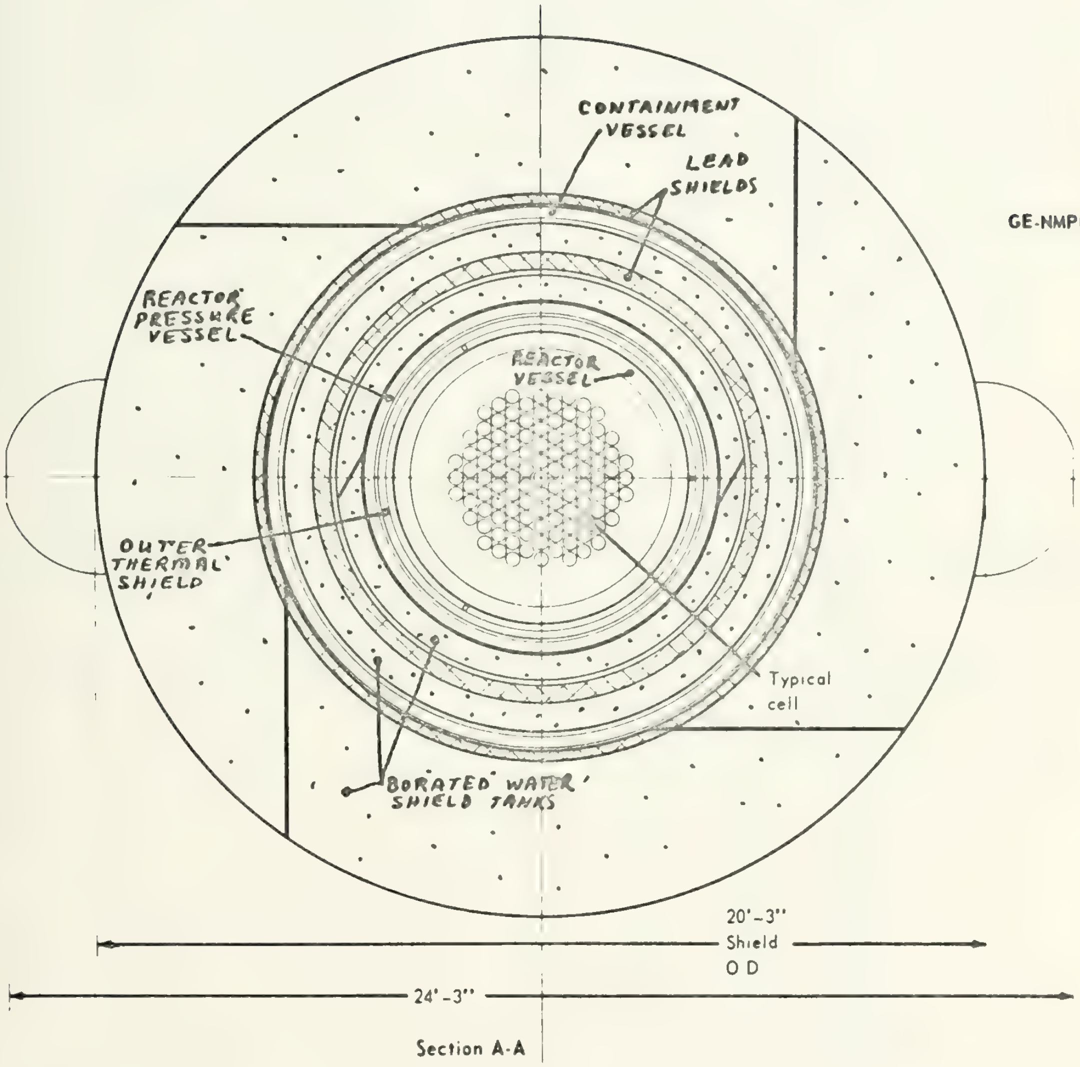


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Figure F-1b 630A Mark V Nuclear Steam Generator Cross Section

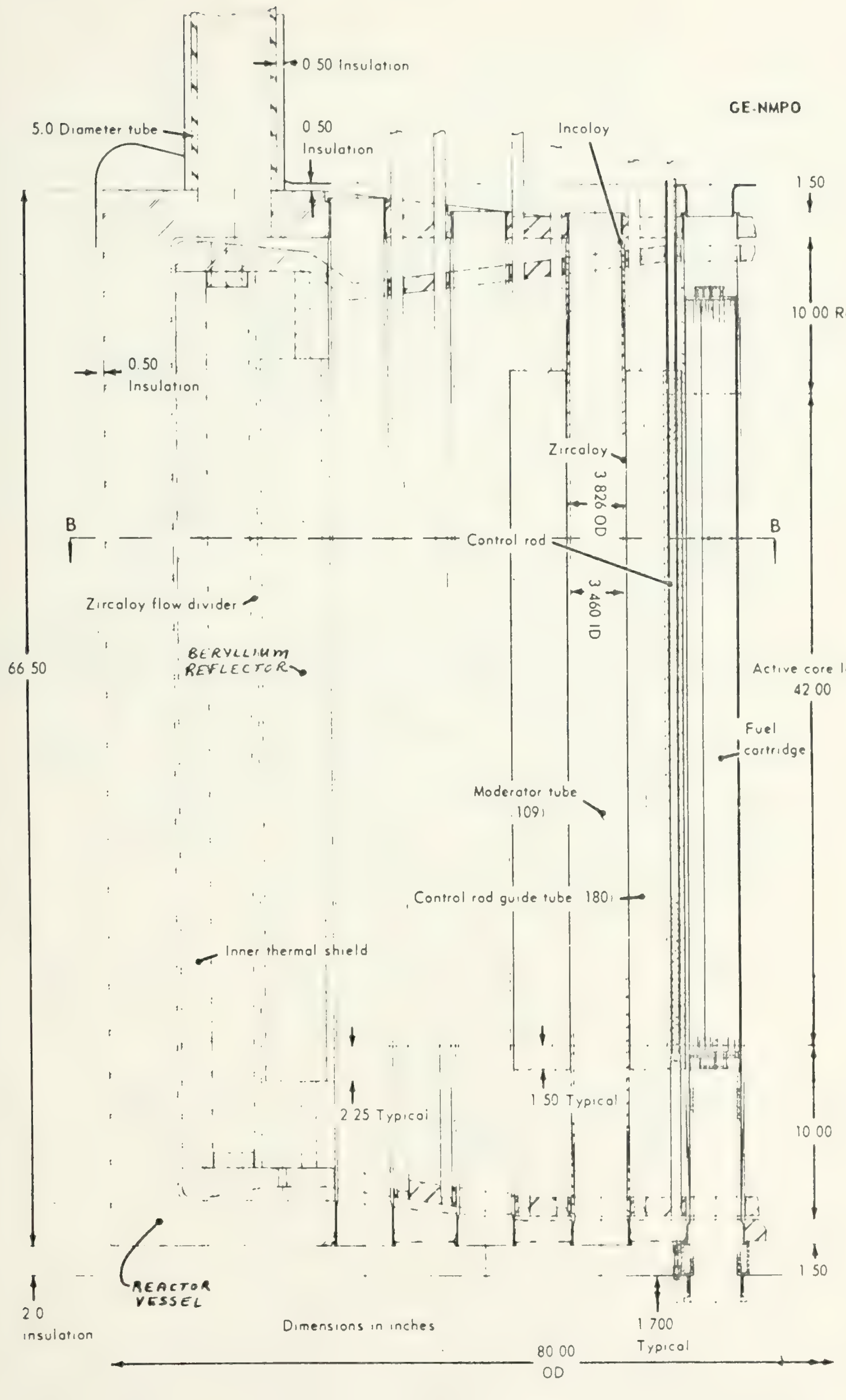


Figure F-2a 630A Mark V Calandria-Type Reactor Design

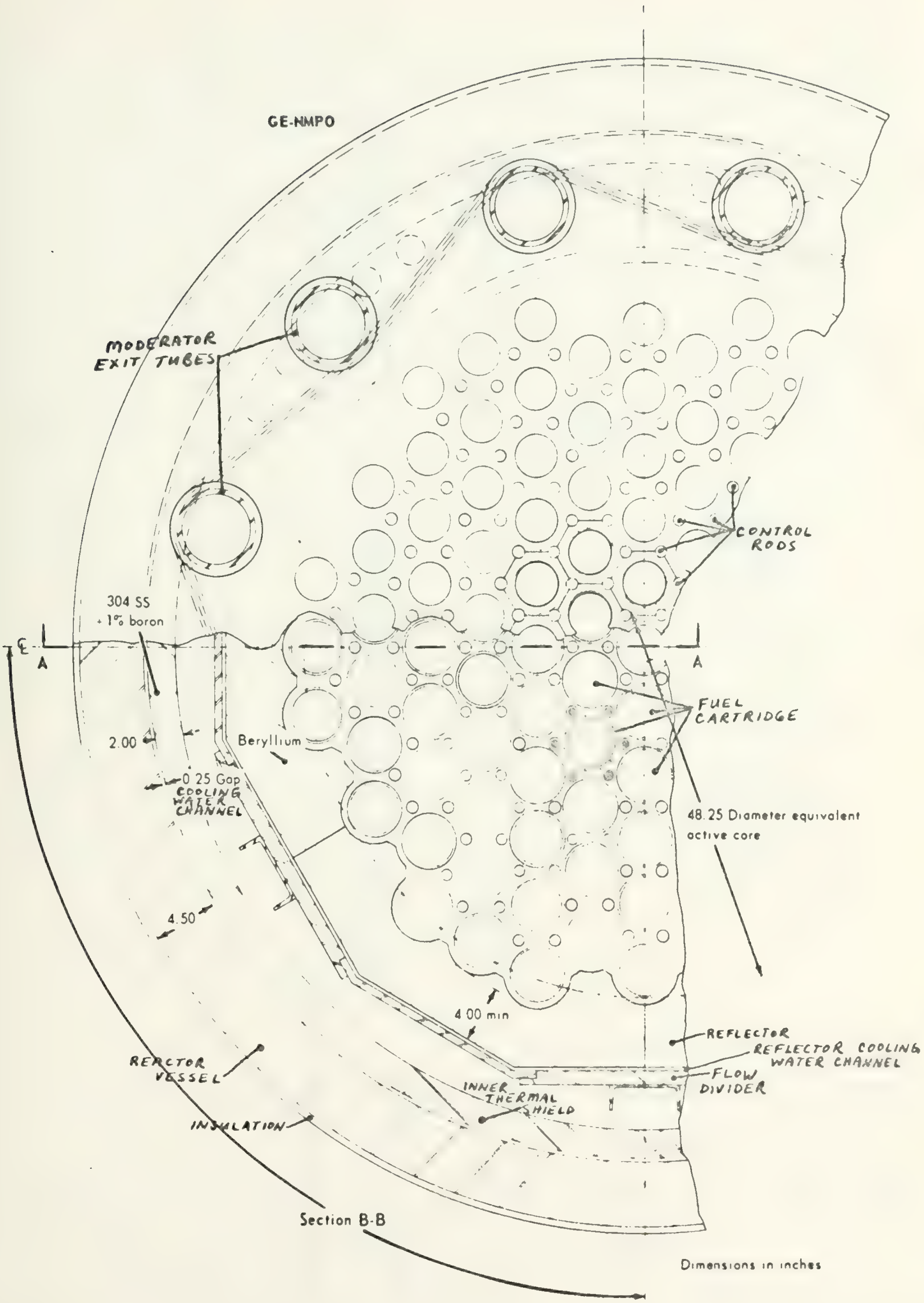


Figure F-2b 630A Mark V Calandria-Type Reactor Design

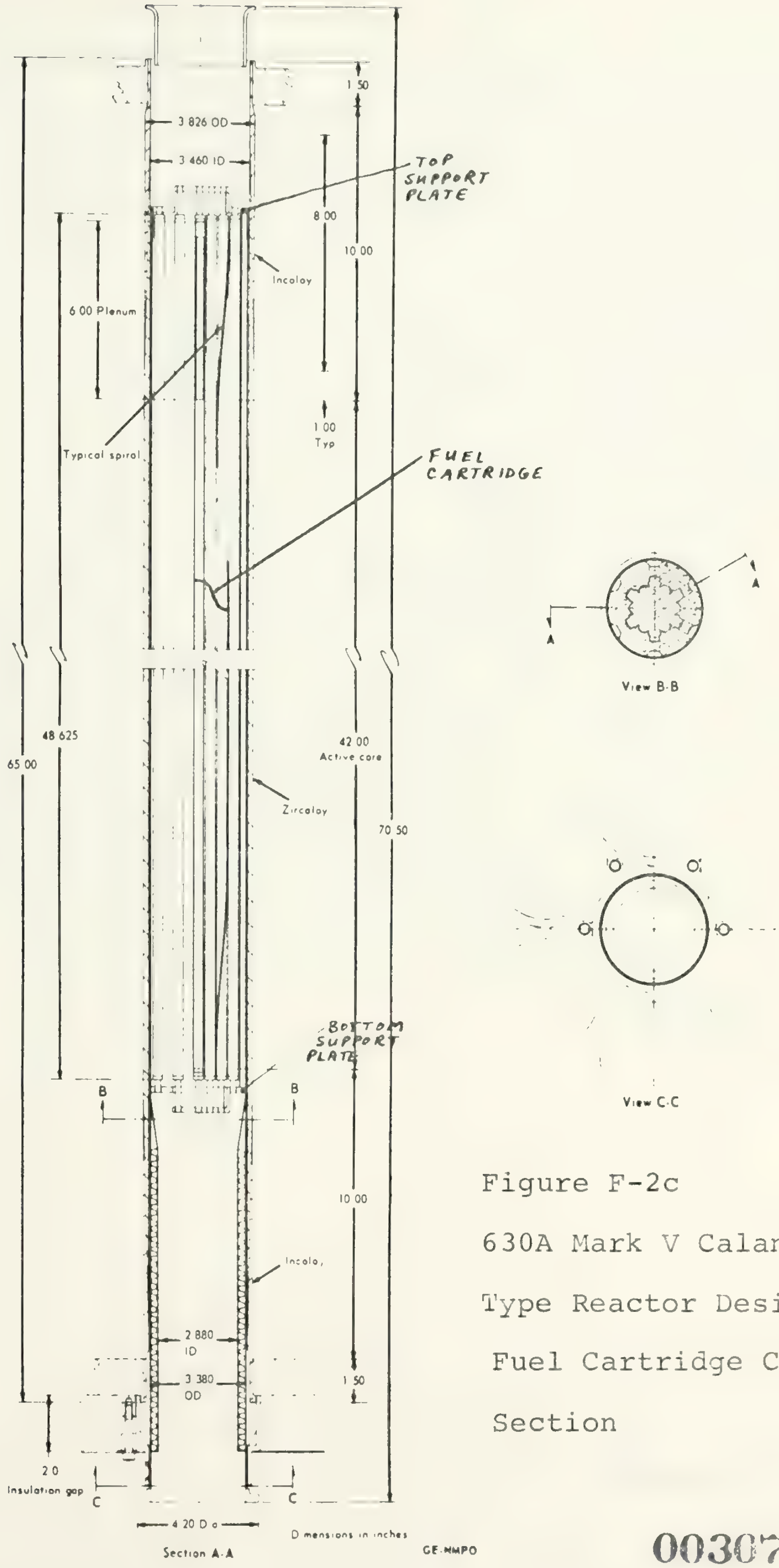


Figure F-2c

630A Mark V Calandria-Type Reactor Design Fuel Cartridge Cross Section

00307

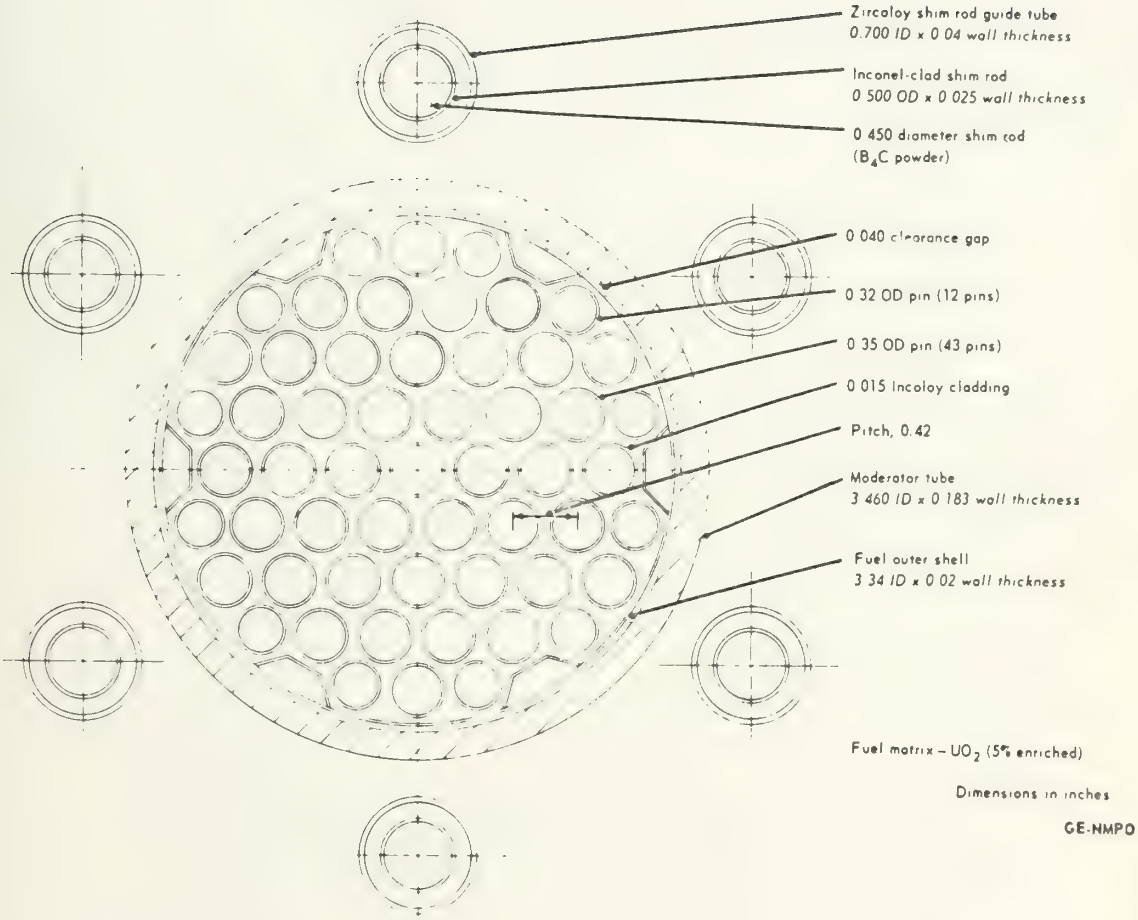


Figure F-2d 630A Mark V Calandria-Type Reactor Design

Fuel Cartridge Cross Section

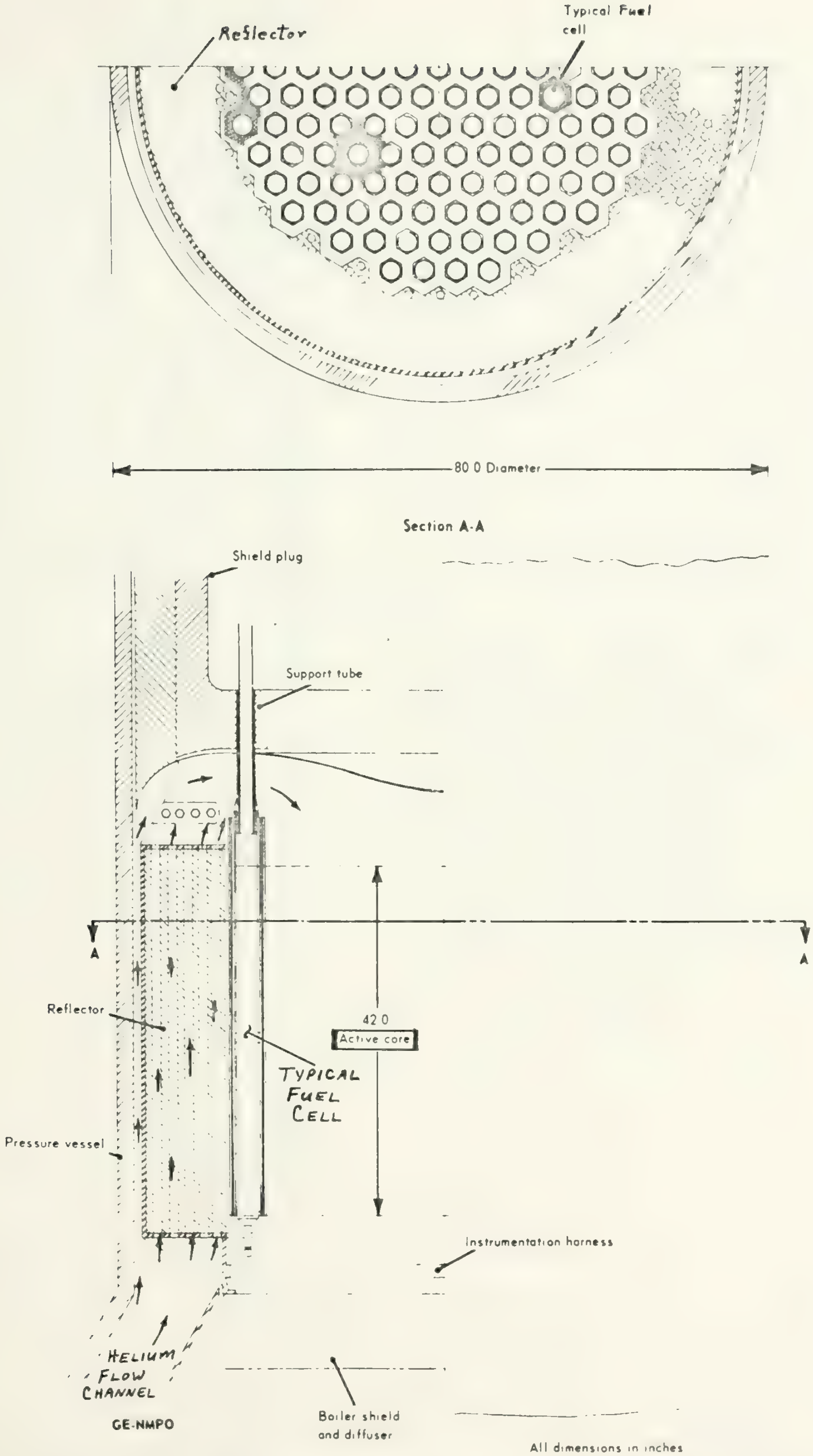


Figure F-3 630A Mark V Tube-Type Reactor Design

00309

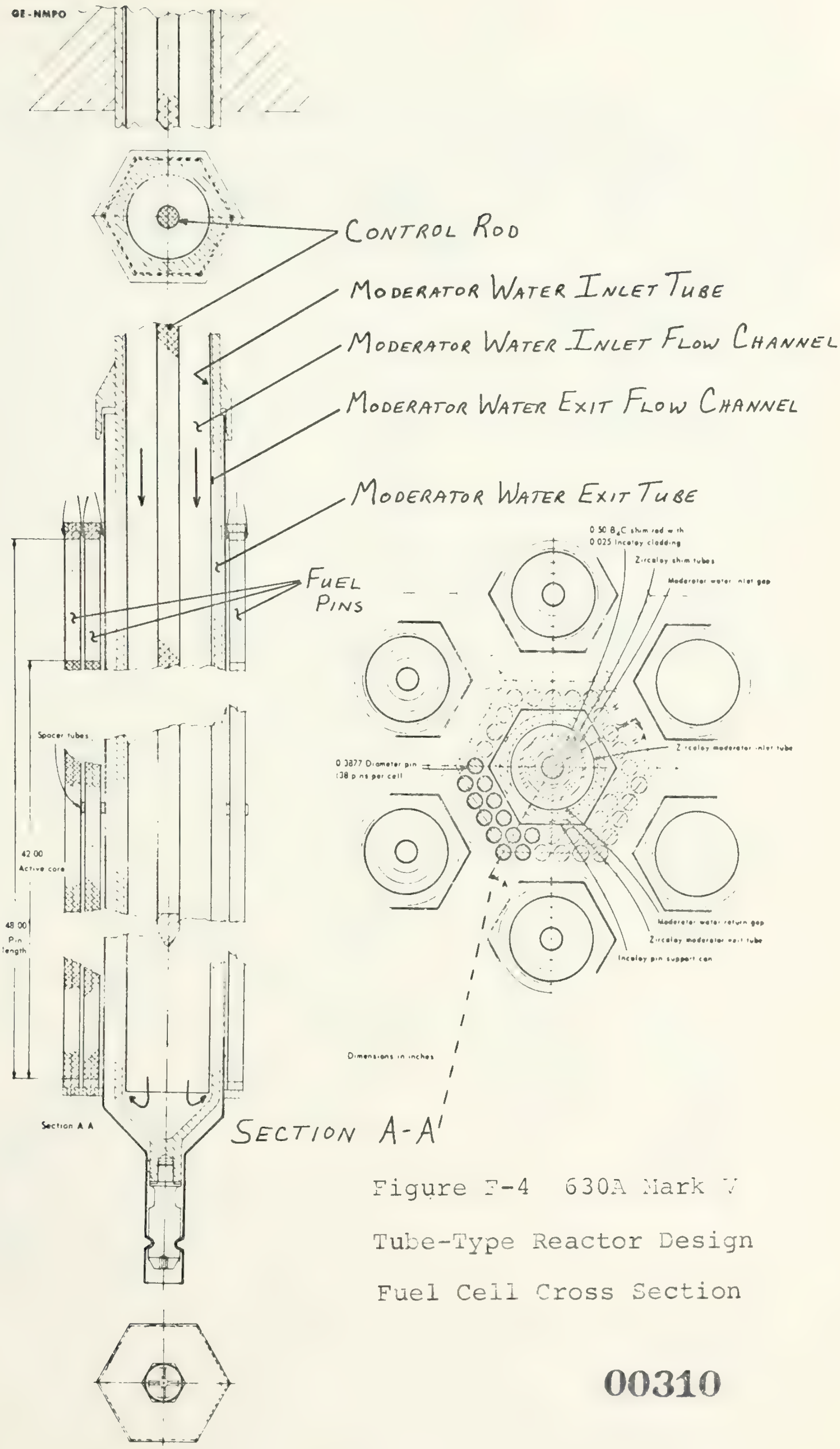


Figure F-4 630A Mark V

Tube-Type Reactor Design

Fuel Cell Cross Section

00310

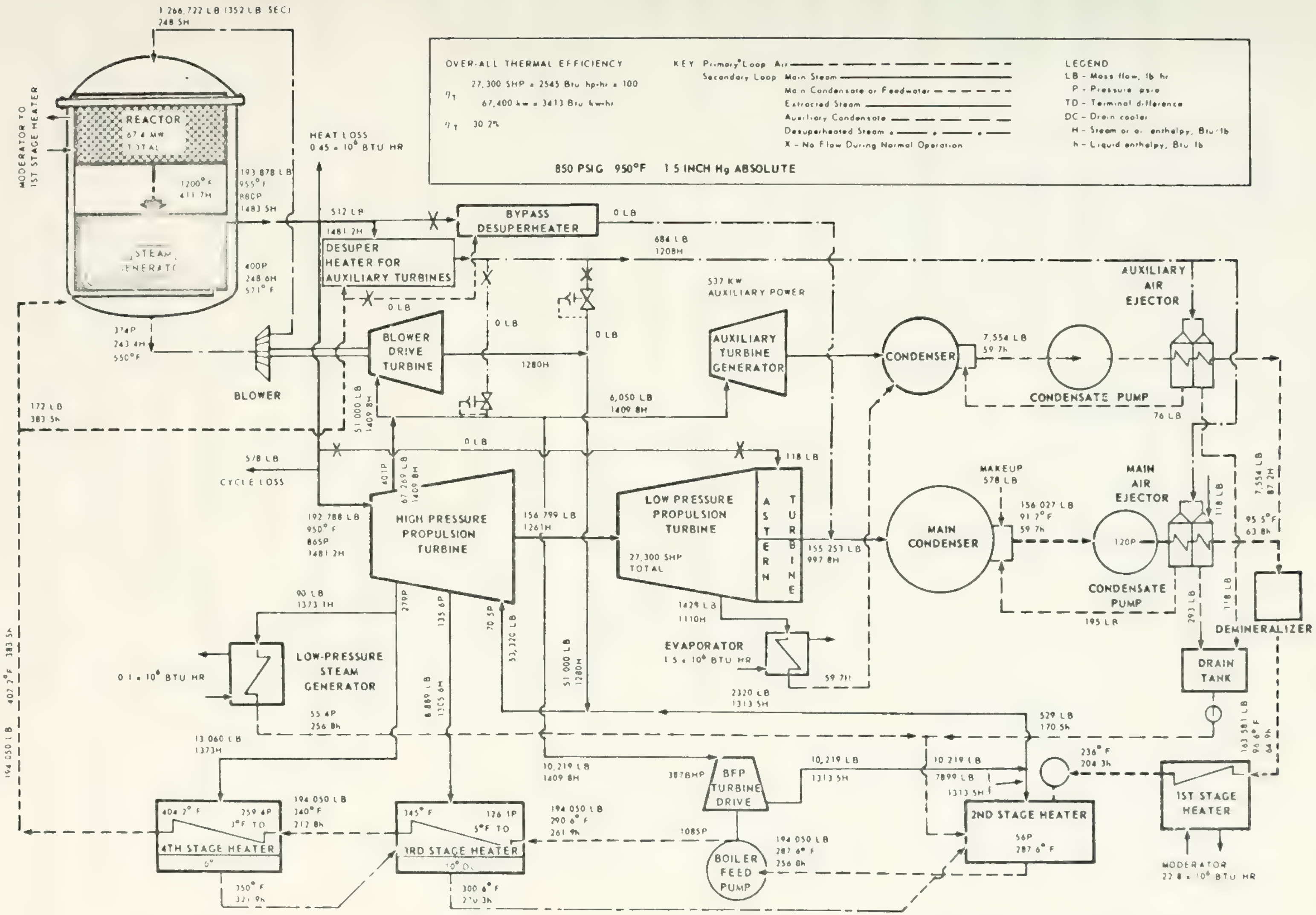


Figure F-5 630A Mark IV Propulsion Plant and Heat Balance Diagram

00311

G. COMBUSTION ENGINEERING'S UNIFIED MODULAR PLANT (UNIMOD) -- (ref's 18, 70, 77, 78, 80, 83)

1. GENERAL --

Like the CNSG, Combustion Engineering's UNIMOD design was developed to provide a compact, low weight, economical, pressurized light water reactor that would occupy less of the ship's total weight and volume than did the SAVANNAH plant. Like the 630A, the UNIMOD design emphasized factory preassembly of the plant into a small number of modules suited for rail shipment and rapid, simple shipboard installation; this feature reduces the ship's time on the building ways and thereby reduces shipbuilding cost. Except for a short run of piping to each primary coolant pump, the UNIMOD design substitutes internal reactor vessel flow passages for the bulky primary system piping of SAVANNAH. A light water cooled and moderated plant like the OTTO HAHN plant, the UNIMOD is self-pressurizing and does not need a separate, bulky pressurizer vessel.

A unique feature of the UNIMOD design is that the reactor is controlled completely through steam demand, with all control rods fully withdrawn from the core, over the entire plant operating power range throughout the core life-time. The current UNIMOD design described below delivers 30,000 SHP, although the basic design is readily adaptable to plants from 10,000 to 60,000 SHP. The containment vessel and its contents weigh 430 tons (325 tons dry, allowing

shipboard installation as a complete, pretested unit), or approximately 32 lbs/SHP for the reactor system; this is not much more than the weight of a comparable oil-fired boiler. Because larger output nuclear plants are not proportionately larger, a 60,000 SHP UNIMOD plant wound be under 40 ft high and 20 ft in diameter and would weigh less than 600 tons.

2. POWER PLANT DESCRIPTION; REACTOR SYSTEM --

a. Containment Vessel --

Figures G-1 and G-2 show a cutaway and a cross section of the reactor containment vessel. This vessel is a vertical cylinder 16 ft OD and 34 ft high, approximately the same size as a comparable oil-fired boiler. The vessel is filled with borated water to a level above the reactor vessel head. This water provides: 1) a reservoir for vapor suppression in the event the primary coolant boundary should rupture, and 2) part of the required radiation shielding (the water volume was determined by shielding requirements rather than vapor suppression). The 1 1/2 in. thick vessel has a design pressure of 300 psig.

b. Radiation Shielding --

Elimination of large primary coolant loops outside the reactor pressure vessel considerably reduces the radiation volume requiring shielding, and, hence, the shield size and weight. The radiation shielding (see Figures G-1 and G-2) consists of the borated water between the reactor vessel and the containment vessel, concentric iron cylinders in this water in the vicinity of the core, plus lead slabs

above and below these cylinders and locally around the 3 short runs of primary coolant piping used. No shielding is installed external to the containment vessel. The radiation shield is designed to allow unlimited access to all compartments outside the containment vessel during reactor operation at full power.

c. <u>Primary System</u> --

As shown in Figure G-2, the light water primary coolant flows upward through the outer 36 fuel assemblies, is redirected downward along the outer periphery of the core by baffles, and flows upward through the central 25 fuel assemblies. Leaving the core at saturation conditions, the heated coolant flows upward to the steam dome and divides radially into the 6 steam generator segments where it makes a downward pass along superheater and steam sections, followed by an upward pass along economizer/downcomer sections. Having traversed the entire length of the steam generator tubing, the coolant recombines into 3 downcomers located between each pair of steam generator segments and leaves the reactor vessel to flow to the 3 main coolant pumps; there it is pumped back into the reactor vessel and directed to the bottom of the core, completing the circuit. The primary coolant pumps are vertical, electric motor driven, centrifugal, single stage, 4300 gpm pumps; any 2 can supply enough flow to obtain full power. Primary coolant flow rate through the core is 12,900 gpm.

At full reactor power of 80 MWt, the core inlet temperature is 610 F and the outlet temperature is 652F. These high temperatures allow use of smaller steam generators than in most other PWR plants. Primary coolant pressure is maintained at the saturation pressure corresponding to core outlet temperature, which is in turn controlled by steam and feedwater flow rates and the inherent reactivity coefficients of the core; this is discussed in greater detail below. The pressurizer vessel and its associated heater and spray control system are entirely eliminated.

The once -- through, counterflow steam generator is arranged along the inside periphery of the reactor vessel. Approximately 17 ft high, it is composed of 6 independent, circumferential segments, any 5 of which will provide the full power steam flow of 300,000 lbs/hr at 600 psig, 600F, 112F superheat. Each segment contains 230, 21.5 ft long parallel tubes from the inlet header to an intermediate header followed by 115, 28 ft long parallel tubes from this header to the outlet header, Half as many tubes are used in the final portion of the steam generators to increase working fluid mass velocity and heat transfer coefficient. All tubes are 1/2 in. OD, 0.085 in. wall thickness, Inconel (a material less susceptible to Cl^- stress corrosion than stainless steel). Total heat transfer area is 6,400 ft^2. Each segment can be isolated on the secondary side with valves external to the containment vessel in the event of major steam generator tube failure. If necessary, the failed segment can be removed and

replaced at the next scheduled ship servicing without removal of the core or other steam generator segments. The core can also be refueled without removal of any of the steam generator segments.

d. Auxiliary Systems --

These systems provide the necessary hydraulic, chemical and heat removal service functions for operation, maintenance and safety of the primary system. Individual systems are combined and simplified where possible to achieve a lighter and smaller overall plant.

e. The Reactor --

i) The Core --

The reactor fuel system consists of 3.93 tons of slightly enriched (average 5.9% with ratio of 1.6 between that in the core outer and inner regions) uranium dioxide pellets contained in stainless steel tubing. The 2-pass core is 42 in. effective diameter and 50 in. high, and contains 61 hexagonal-shaped fuel assemblies; 12 of these assemblies are moveable axially. Each fuel assembly contains 126 (127 in moveable assemblies) 0.328 in. OD, 0.015 in. wall thickness stainless steel tubes, most of which contain UO_2 pellets; a few of these tubes contain boron carbide burnable poison to minimize reactivity variation over core life. The 12 cylindrical, water-filled, stainless steel clad, boron carbide control rods are surmounted on the 12 movable fuel assemblies and symmetrically located in the inner, second pass region of the core. These rods are fully withdrawn

during power operation, raising the 12 fuel assemblies into their normal position in the core, and are inserted only to achieve prolonged reactor shutdown or reactor scram protection. Rod insertion removes these 12 fuel assemblies almost completely from the core; the upper six inches of these assemblies remains inside the core boundary to help reduce the required reactor vessel height. Cold shutdown margin is greater than 7% $\Delta k/k$.

Radial neutron flux and power distribution flattening are achieved by the use of 2 fuel enrichment zones and the 2 pass core (the latter feature also reduces the required coolant pumping power). The higher peripheral enrichment and lower moderator temperature both tend to raise the neutron flux and power in this region of the core.

Burnable poison is used to minimize lifetime reactivity variation. The core has a design fuel burnup of 20,000 MWD/metric ton of uranium, yielding a core life of 3.4 years at an 80% usage factor (23,000 equivalent full power hours). In a complete loss of flow accident the saturated moderator reduces power rapidly by temporarily boiling in the core, so that the core is protected against this accident without scram. Moderator to fuel volume ratio is 1.35, chosen as a compromise to give acceptable values of inherent reactivity control, thermal-hydraulic characteristics, average fuel enrichment, core life, and core size. The temperature coefficient of reactivity is $-1 \times 10^{-3} \Delta k/k/F$.

Maximum linear power generation rate in the fuel rods is 7.5 kw/ft at 125% power.

ii) The Reactor Vessel --

The SA-302B low alloy steel reactor vessel is a 71 in. ID x 22 ft 4 in. high vertical cylinder with welded ellipsoidal bottom head and bolted hemispherical top head. Internal surfaces are clad with austenitic stainless steel to reduce corrosion. The vessel is fully insulated and encapsulated, including primary coolant pumps and piping and control rod drive mechanisms, with watertight canning. Vessel design pressure is 2800 psig at 685F.

f. Reactor Control System --

By more fully utilizing the inherent load-following characteristics of a water-cooled reactor than has been done in other plants, the UNIMOD design totally eliminates the requirement for direct reactivity control during operation. The resulting simpler reactor plant control is provided via steam demand as follows:

- Steam pressure is maintained in a narrow band by regulating feedwater flow via a variable speed pump.

- Primary coolant core inlet temperature is determined, for constant coolant flow rate, by steam flow rate.

- Reactor power level is determined by primary coolant core inlet temperature via the core's negative temperature coefficient.

- Primary coolant outlet temperature is determined,

for constant coolant flow rate, by core inlet temperature and reactor power level.

- Primary coolant pressure is determined by primary coolant core outlet temperature.

Results of detailed calculations and analog computer simulations indicate the following ranges of plant steady state behavior over core life; these are displayed graphically in Figure G-3:

- Primary coolant pressure varies as follows:
  - no-load to full power, 180 psi increase
  - beginning of life to end of life, 340 psi variation
  - no xenon to full power equilibrium xenon concentration in core, 200 psi decrease
  - total range of pressure variation, 2000 to 2500 psi
- Turbine throttle steam temperature varies between 600F and 650F for all conditions during the life of the core.

The UNIMOD design can also respond rapidly and safely to rapid changes in steam demand, due to the stabilizing effect of the core's inherent negative temperature coefficient, the Doppler coefficient of its uranium based fuel system, and its negative void (from boiling in the core) coefficient. For example, Figure G-4 shows analog computer calculated predictions of plant response to step changes in steam demand from 100% to 50% and return.

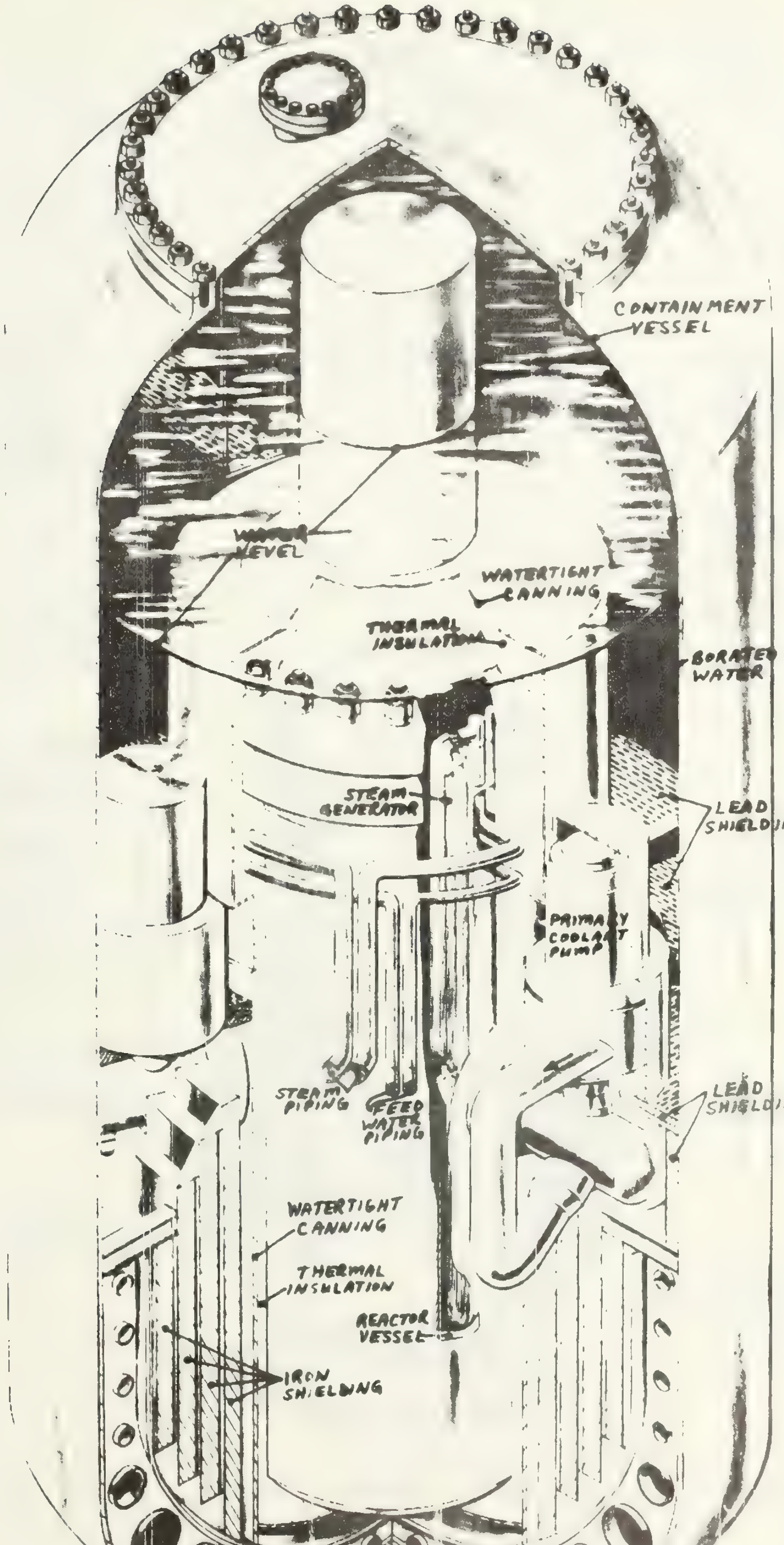


Figure G-1 UNIMOD Reactor Plant Arrangement

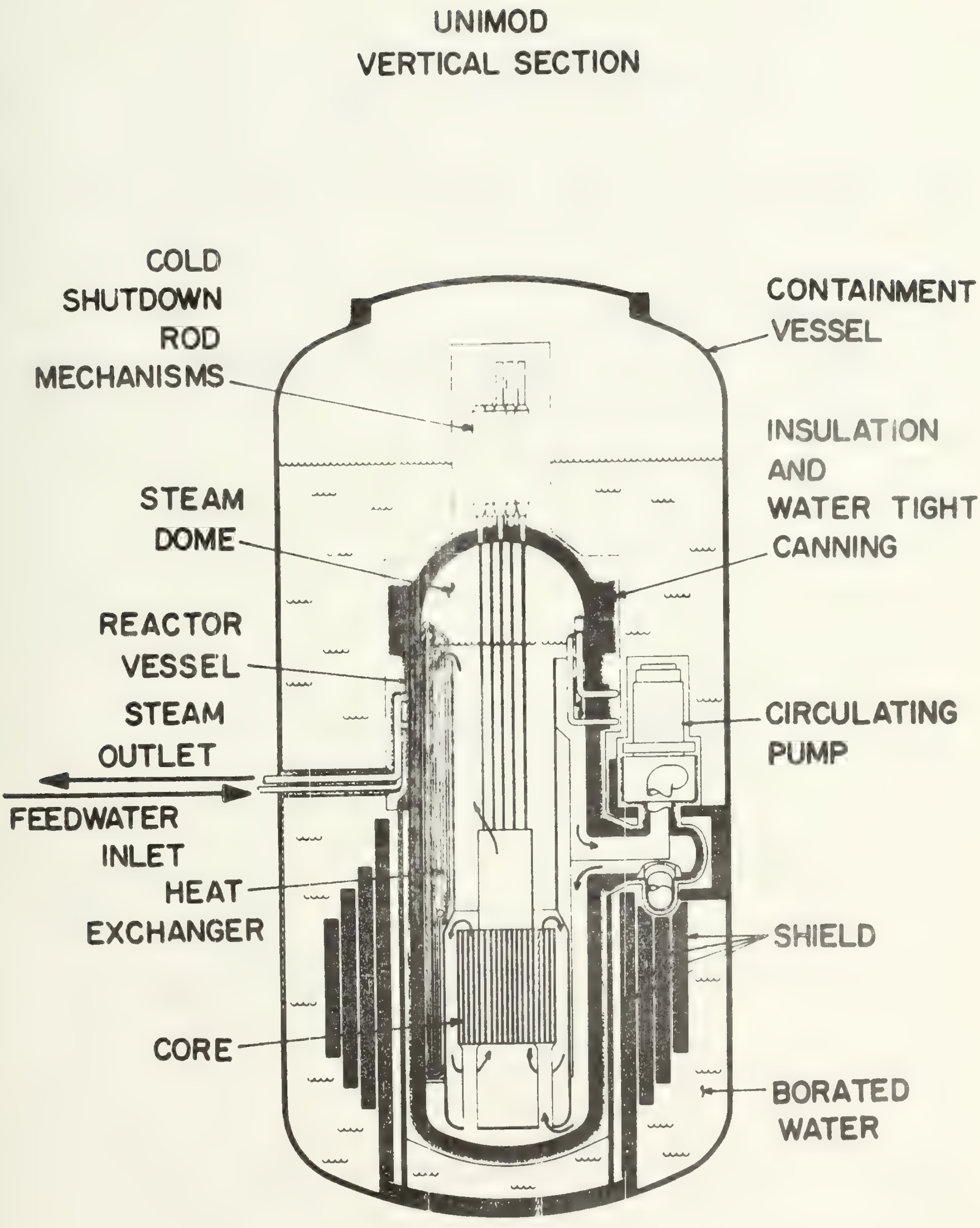


Figure G-2 UNIMOD Reactor Plant Vertical Cross Section

UNIMOD
REACTOR LOAD CHARACTERISTICS

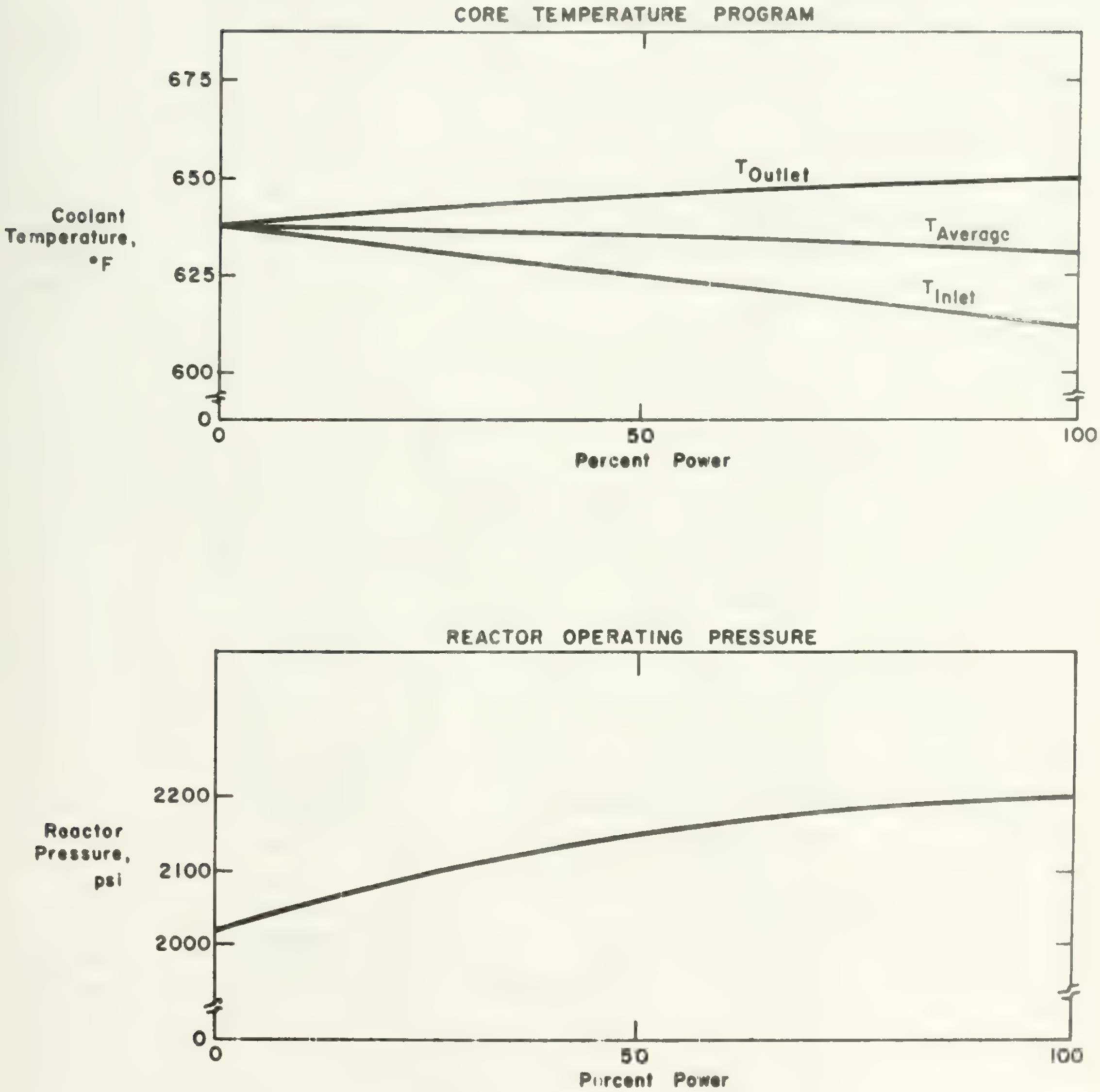


Figure G-3a UNIMOD Reactor Load Following Characteristics

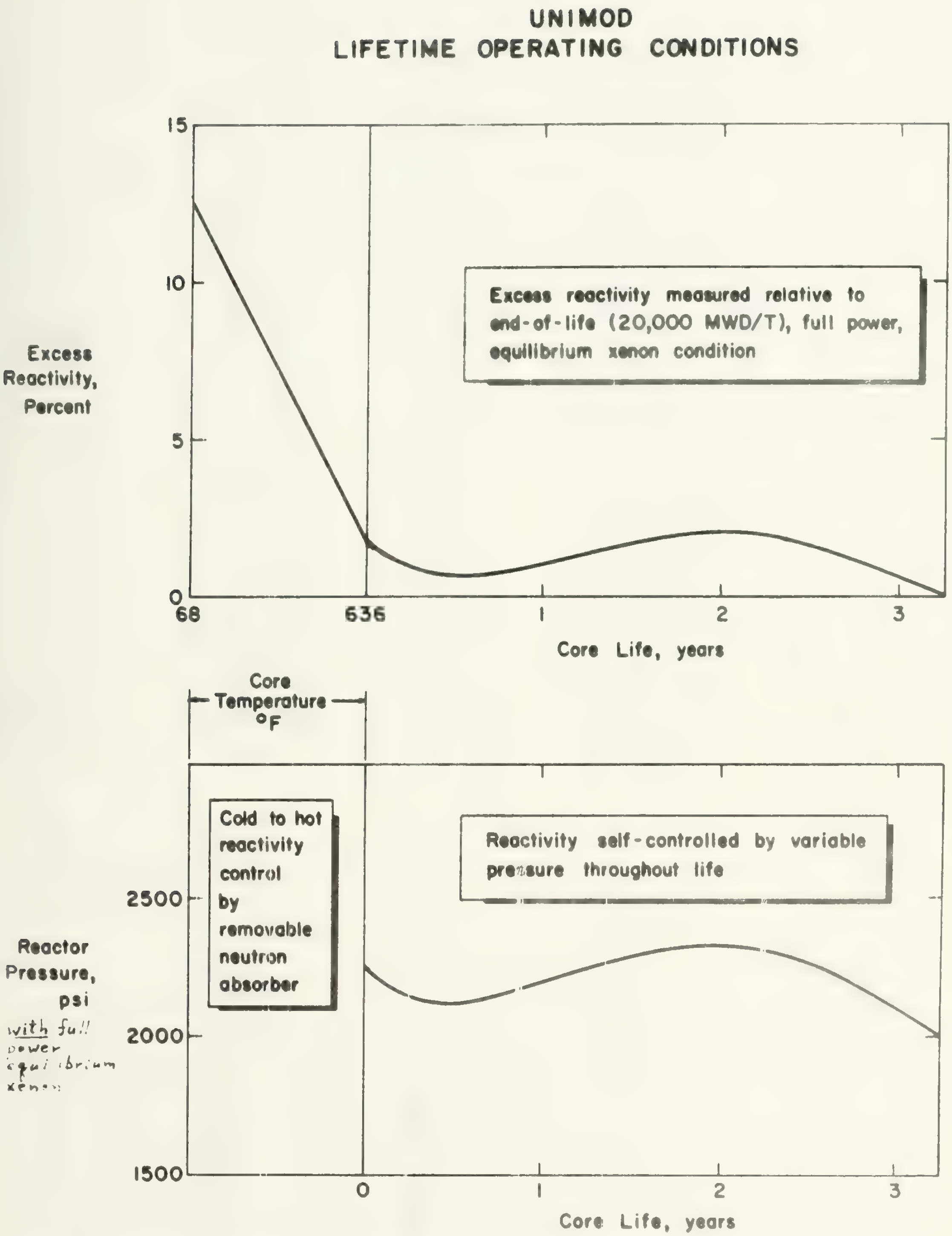


Figure G-3b UNIMOD Reactor Plant Lifetime Pressure and Reactivity Characteristics

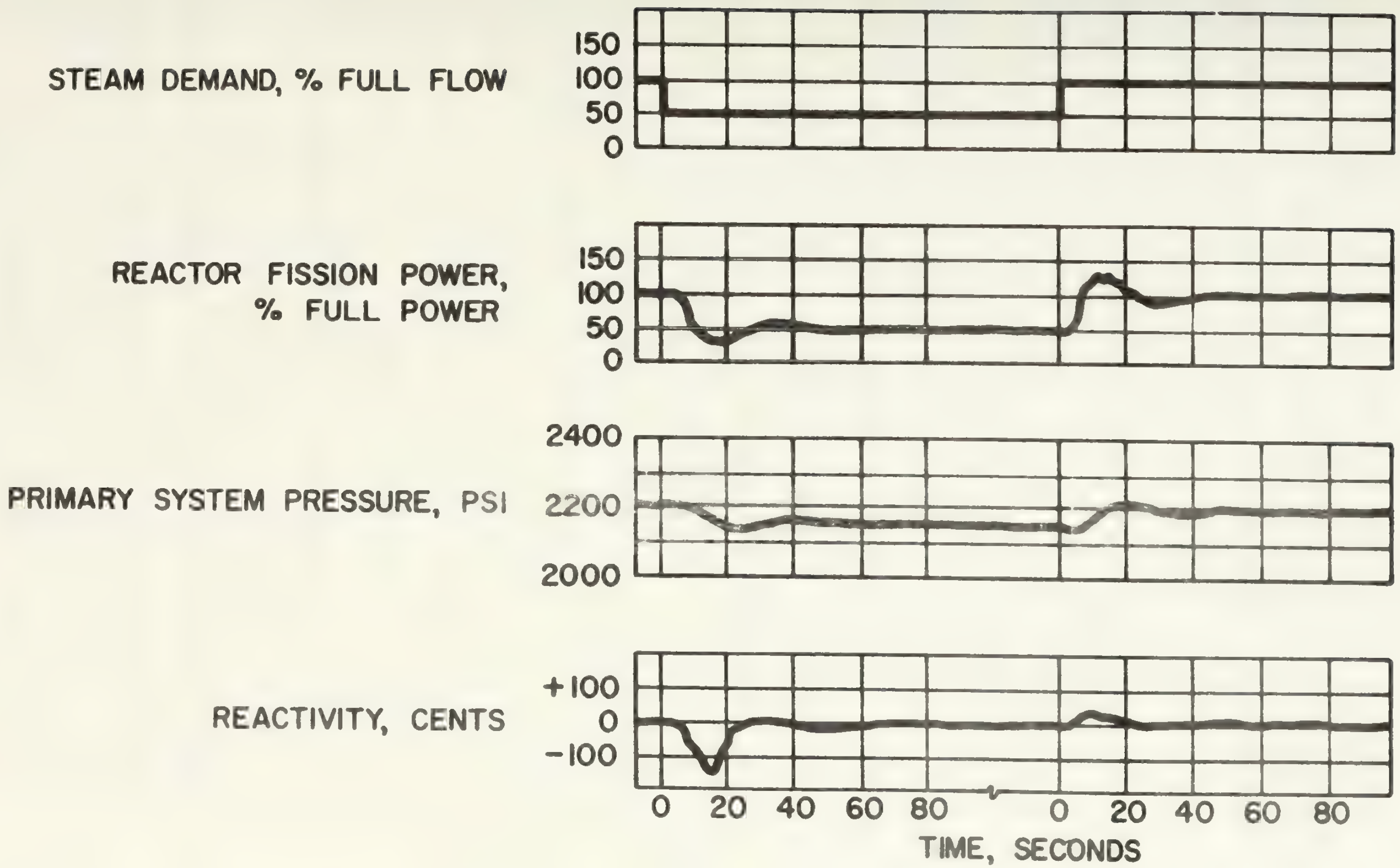


Figure G-4 UNIMOD Reactor Plant Response to Steam Demand Changes

(Analog Computer Simulation Predictions)

Appendix II

SURVEY OF AUTHORITIES IN THE NUCLEAR AND MARINE ENGINEERING FIELDS

In order to obtain the views of authorities in the nuclear and marine engineering fields regarding certain aspects of commercial nuclear marine propulsion, a short questionnaire consisting of 7 questions was designed to provide the information desired with a small expenditure of time on the part of each person surveyed.

For the most part, the questionnaires were addressed to company presidents, department heads, etc. by name, rather than to firms or agencies. The individuals surveyed are divided into 6 categories depending on the nature of the firm or agency with which each is associated. Although it is understood that the views of an individual do not necessarily represent the views of his firm or agency, it is felt that his views are enough influenced by the nature of his work that it might be useful to have the authorities divided into these categories. The categories are as follows; all 152 individuals surveyed are associated with U.S.-based firms or agencies:

A. The Academic Community -- individuals associated with departments of nuclear engineering, naval architecture and marine engineering, and ocean engineering in colleges, universities, and maritime academies; 26 responded.

B. The Ship Building Community -- individuals associated with large shipyards which either have or could

conceivably have nuclear capability; of the 7 who responded, 3 declined specific comments based on insufficient current applicable knowledge.

C. The Ship Owner/Operator Community -- individuals associated with large maritime shipping firms; of the 12 who responded, 2 declined specific comments based on insufficient current applicable knowledge.

D. The Reactor/Reactor Equipment Manufacturing Community -- individuals associated with firms engaged in the manufacture of reactors or major reactor plant equipment; of the 18 who responded, 2 declined specific comments based on insufficient current applicable knowledge.

E. The Marine/Nuclear Consulting Community -- individuals associated with consulting firms in the marine and nuclear engineering fields; of the 17 who responded, 3 declined specific comments based or insufficient current applicable knowledge.

F. Government -- individuals associated with government agencies involved in commercial nuclear marine propulsion; 3 responded.

The results of the survey, including the many comments received, are presented below.

<u>Question 1</u>. What type(s) commercial ships do you consider most suitable for nuclear propulsion? Please number, to whatever extent you may desire, in order of decreasing suitability.

General Cargo Ship______

Containership__________

Tanker__________________

Ore Carrier_____________

Passenger Liner_________

Composite Ship__________

Other (please identify)______________

If you care to comment further, what is the basis for your selection? (e.g., long routes, high speed, fast turnaround, built-in collision protection, etc.)

The 6 histograms in Figure App II-1 below present the ranking of suitability of the first 6 ship types mentioned in this question. Other ships selected and the ranking for each ship type are as follows:

a) Icebreakers - 1st choice twice, 3rd once, and 4th three times

b) LASH Ship - 1st choice twice

c) Fishing Vessel - 3rd choice once

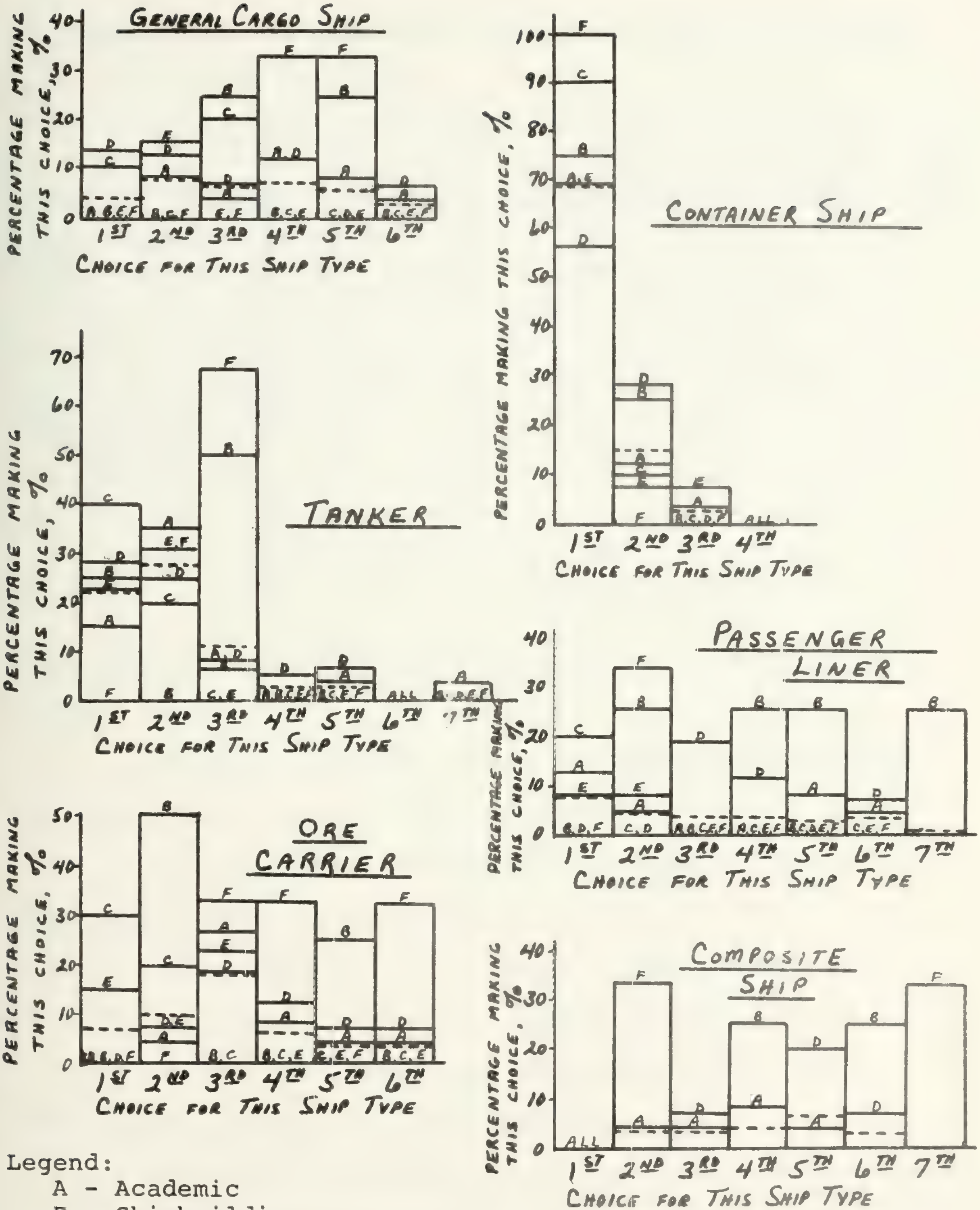
d) Roll on/roll off Barge Carrier - 1st choice once

e) Long Run International Ferries - 5th choice once

f) Surface Effect Ships - 4th choice once

g) Commercial Cargo Submarines - 5th choice twice

The bases stated for the selection of this ranking of ship type suitability are as follows, where A through F refer to the categories of individuals as listed above and



Legend:
- A - Academic
- B - Shipbuilding
- C - Owner/Operator
- D - Manufacturing
- E - Consultants
- F - Government
- ----Combined

Figure App II-1 Responses to Survey Question No. 1

the numbers are the percentage of these individuals giving this basis for selection:

Basis	A	B	C	D	E	F	Combined
High Speed/ High Power	69%	75%	60%	50%	77%	67%	66%
Long Routes	46%	50%	50%	37%	38%	33%	42%
Fast Turnaround/ High Ship Utilization	65%	50%	10%	44%	38%	67%	47%
No Comment	19%	75%	30%	40%	15%	0%	27%

Other bases and comments given (and their frequency) are as follows:

a) Fast air travel has resulted in low demand for even high speed passenger liners (once).

b) Air cushion vehicles offer possibilities if power plant weight can be reduced (once).

c) Nuclear is well suited for long periods of operation in remote regions like Arctic or Antarctic (once).

d) High non-propulsion ship capital cost in a tanker could balance out higher nuclear plant cost (once).

e) Large tonnage favors nuclear economics (4 times).

f) High value cargo in containership gets a premium for prompt delivery (4 times).

g) Tankers are too dangerous due to poor weight/

strength ratio and susceptibility to collision (once).

h) Tug-barge concept warrants more study; may have biggest payoff (once).

<u>Question 2.</u> For what SHP range would you expect nuclear propulsion to be economically competitive with more conventional propulsion for commercial ships?

Service Application		SHP Range <10,000	10-50,000	50-100,000	>100,000
<u>By 1980</u>	A	4%	15%	19%	35%
	B	--	--	75%	25%
	C	--	--	10%	50%
	D	--	13%	13%	63%
	E	--	--	57%	36%
	F	--	--	33%	100%
	Comb.	1%	8%	27%	45%
<u>By 1990</u>	A	4%	8%	35%	23%
	B	--	--	--	50%
	C	--	--	20%	20%
	D	--	--	44%	25%
	E	--	29%	29%	7%
	F	--	--	100%	--
	Comb.	1%	8%	34%	21%
<u>By 2000</u>	A	4%	15%	39%	35%
	B	--	50%	25%	--
	C	--	20%	30%	20%
	D	--	6%	6%	19%

	E	14%	21%	14%	7%
	F	--	100%	--	--
	Comb.	4%	21%	23%	21%
<u>Never</u>	A	31%	8%	--	--
	B	75%	--	--	--
	C	20%	10%	--	--
	D	19%	12%	--	--
	E	21%	7%	--	--
	F	67%	33%	--	--
	Comb.	29%	10%	--	--

Comments made on question 2 are as follows:

a) SHP's above 100,000 are economically competitive today.

b) "Economically competitive" requires subsidy. Economic assumptions must be stated in order to give a meaningful answer.

<u>Question 3</u>. What reactor type(s) do you consider presently most suitable for commercial ship application? Please number, to whatever extent you may desire, in order of decreasing suitability.

Pressurized Water Reactor (PWR) (loop type) ______

PWR (integral or consolidated type) ______

Boiling Water Reactor ______

Gas Cooled Reactor (steam turbine) ______

Gas Cooled Reactor (indirect cycle gas turbine) ______

Gas Cooled Reactor (direct cycle gas turbine) ______

Liquid Metal Reactor (steam turbine) ________________

Liquid Metal Reactor (gas turbine) ________________

Organic Cooled and Moderated Reactor (steam turbine) ___

Other (please identify) ________________________

If you would care to comment further, what is the basis for your selection?

The 9 histograms in Figure App II-2 below present the ranking of suitability of the first 9 reactor types mentioned in this question. Other reactor types selected and the ranking for each reactor type are as follows:

a) UO_2 fueled, hydride moderated BWR - 1st choice once

b) Molten salt reactor - 3rd choice twice

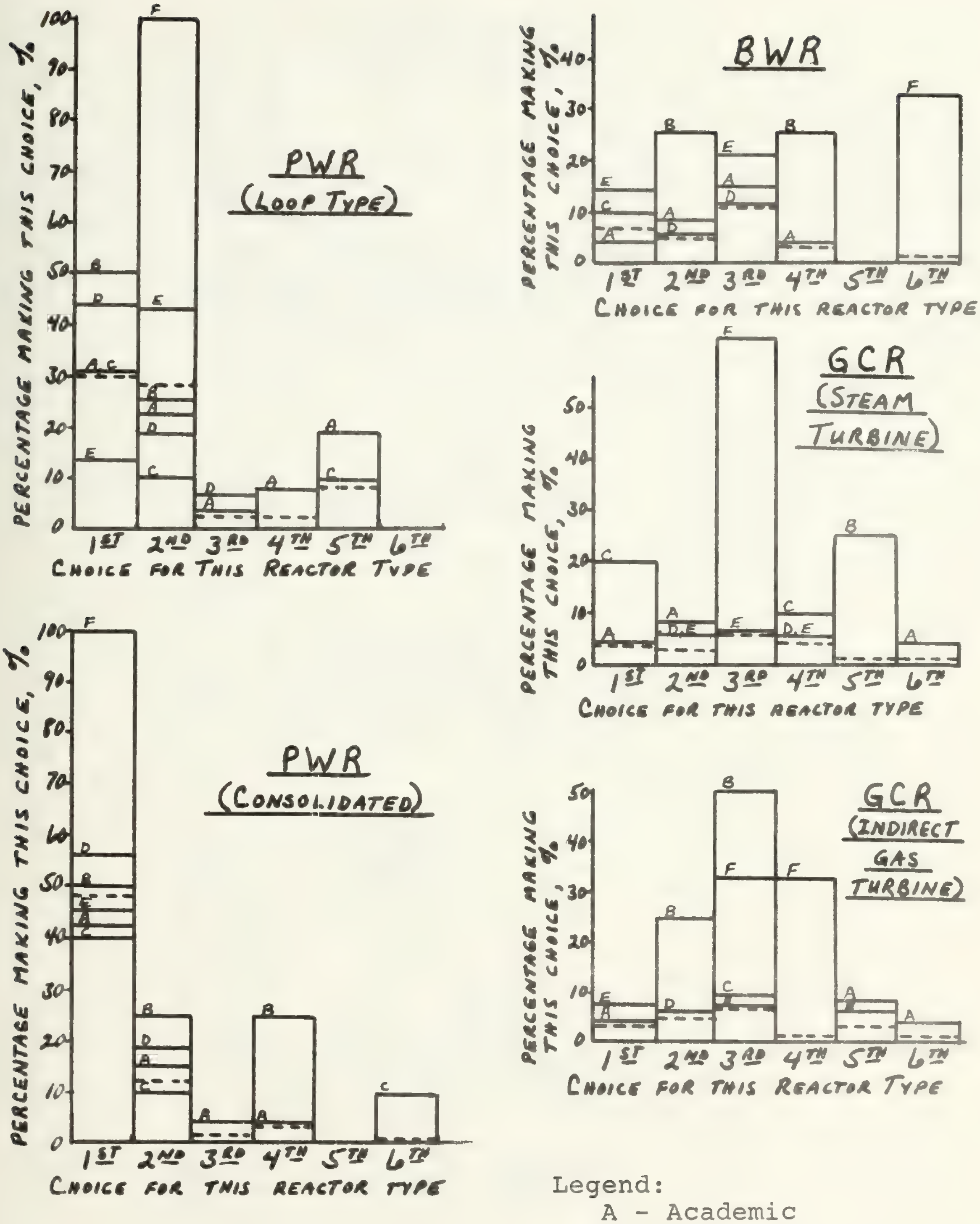
c) Supercritical water reactor (steam turbine) - 6th choice once

The bases stated for the selection of this ranking of reactor type suitability are as follows; where numbers in parentheses indicate the number of times, if more than one, that basis was stated:

Academic --

PWR -- proven (3); safe; minimum cost; simple to operate; it is doubtful that any acceptable amount of development could bring any other type to parity.

PWR (loop type) -- extensive Navy experience (2); here today; proven; reliable



Legend:
A - Academic
B - Shipbuilding
C - Owner/Operator
D - Manufacturing
E - Consultants
F - Government
----Combined

Figure App II-2 Responses to Survey Question No. 3

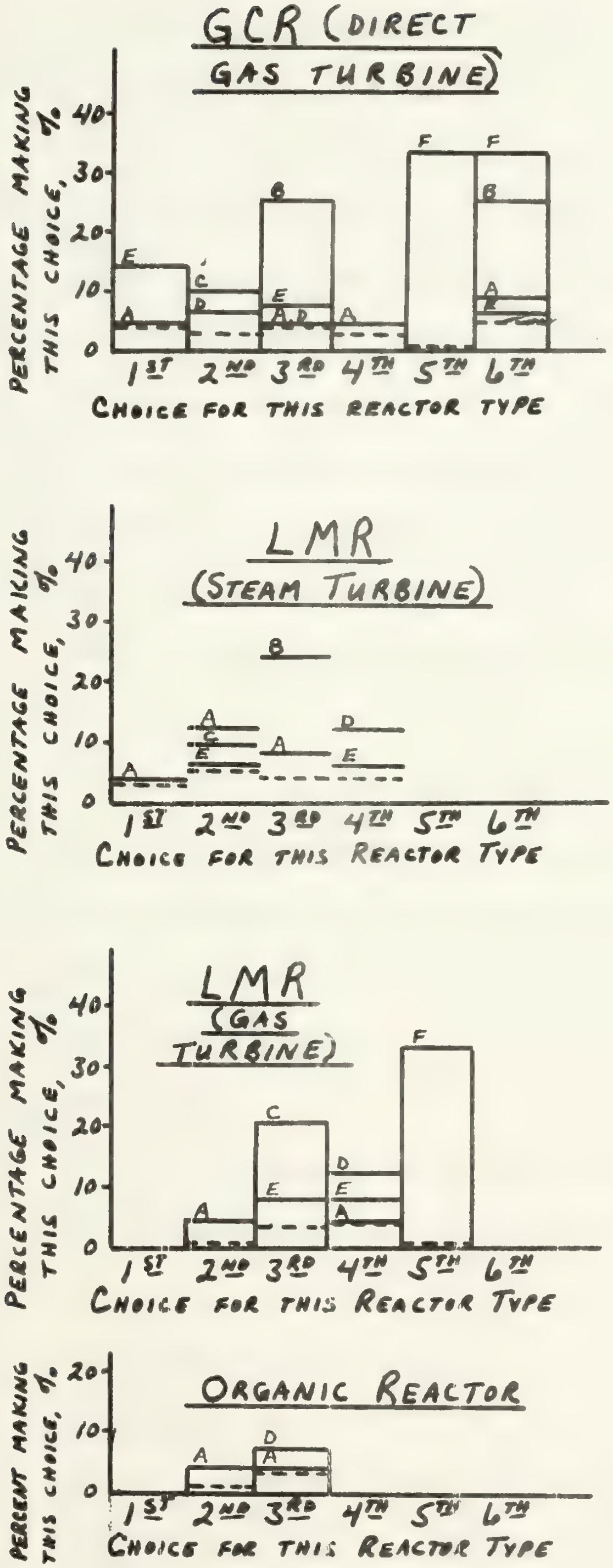


Figure App II-2 Responses to Survey Question No. 3 (Cont'd)

PWR (consolidated) -- proven design; compact plant (2); low weight

GCR -- low size and weight, so low cost; too large due to low heat transfer and material temperature limitations; minimum cost & maintenance & operators; maximum reliability & safety & auto control

LMR -- compact and high steam temperature (2); too hot for reliable low cost; primary coolant activation too high; gas turbine won't compete with steam; not much justification for liquid Na for marine use

OMR -- not suitable due to large radioactive storage

General -- a maritime reactor must be a parasite on central station technology to be economically viable; it's hard to support direct cycles for ship application because of increased safety problems.

No Comment (10)

<u>Shipbuilding</u> --

PWR -- maximum use of existing technology (3); a minimum of development must be involved if to be truly commercial.

No Comment (1)

<u>Owner/Operator</u> --

PWR -- current state of the art (3)

PWR (loop type) -- the only one with significant shipboard experience

PWR (consolidated) -- relatively low weight

and space requirements; high degree of proven reliability

OMR -- offers prospect of light weight & low operating pressure.

Manufacturing --

PWR -- I would not go to sea with anything but a PWR; proven in Navy program (2); others not suitable now or in near future; more reliable; less radioactivity release (2); safer; smaller; easier to maintain

PWR (loop type) -- nuclear Navy designs are most suitable because of experience curve considerations (2).

PWR (consolidated) -- the only type being offered commercially today, to my knowledge

GCR -- direct cycle gas turbine plant could provide high performance relatively soon; a plant with a He gas turbine will be a positive contender for lightweight, advanced concept vehicles (e.g. Surface Effect Ships in 10,000 ton range).

General -- must use a fuel cycle supportable by stationary nuclear fuel fabricators; marine nuclear fuel requirements will be so small for so long that they cannot independently support a separate, widely different, marine fuel processing industry; types other than PWR and OMR require either too much space or too high fuel cost or fail on reasonable safety considerations.

No comment (6)

Consultants --

PWR -- experience has proven the design (4); reliability - commerce functions on reliability & labor accord above all other considerations; stability

PWR (consolidated) -- weight, space and cost savings

No comment (9)

Government --

PWR -- reliability; availability of off-the-shelf components and technology (2); OTTO HAHN experience

Question 4. Part a) Which of these reactor types (if any) do you consider should receive further research and development for commercial nuclear propulsion application? If you would like to comment further, what is the basis for your selection?

The selections, numbers of individuals making them in parentheses, and bases for the selections, if any, are as follows:

Academic --

PWR (loop type) (3) -- most probable to give high return for development money spent; less stringent safety requirements and overdesign

PWR (consolidated) (6) -- adaptability to existing hull designs; most probable return for development dollar

BWR (1)

GCR (steam turbine) (5) -- high temperatures

give small size and weight/SHP; should use indirect cycle to keep as many barriers between the reactor and the general public and the operators as possible.

GCR (direct cycle) (3)

LMR (steam turbine) (3) -- could reduce operating cost if a breeder, but need to know reactor dynamics.

OMR (1) -- maximum safety, minimum cost, minimum complexity.

Molten salt fueled (1)

UO_2 fueled, fixed hydride moderator, 1200-1400 psi BWR, turbo pump feed & circulation (1) - to minimize reactivity swing hot to cold while maintaining a compact power system & compact shielding & high performance.

None (3) -- use technology developed for central stations; use R&D money for breeder reactors.

The improvement to be expected from a liquid metal or molten salt reactor is not enough to make the difference between success and failure.

No comment (8)

Shipbuilding --

BWR (2)

GCR (2)

GCR (indirect, gas turbine) (1)

LMR (steam turbine) (1)

Owner/Operator --

PWR (consolidated) (2)

GCR (2)

GCR (indirect, gas turbine) (3) -- possible weight savings; major savings in support equipment; reliability

LMR (1)

OMR (1) -- light weight & low operating pressure

Government should just build nuclear ships; development will follow from experience.

No comment (2)

Manufacturing --

PWR (loop type) (2) -- should be based on naval and central station technology.

PWR (consolidated) (4) -- not as much R&D involved; potential for high reliability & reasonable size.

GCR (2)

GCR (direct, gas turbine) (2) -- 35-40% efficiency can be attained with state of the art technology and only engineering development; a very compact unit could make significant savings through optimized ship system design.

NERVA adaptation for gas turbine applications (1)

LMR (1) -- apply breeder technology

OMR (1) -- possible capital cost reduction

None (5) -- main problems are economic rather

than technical; nuclear propulsion for ships will always be a marginal proposition at best; better to put R&D $'s to ship systems to which nuclear power could be economically applied; nuclear plant technology is already well defined.

No comment (2)

<u>Consultants</u> --

PWR (loop type) (1)

PWR (consolidated) (1)

BWR (1)

GCR (direct, gas turbine) (1) -- efficiency

Pu fuel system (1)

None (1)

No comment (8)

<u>Government</u> --

PWR (consolidated) (2)

GCR (2)

<u>Question 4. Part b)</u> Who do you consider should sponsor this development effort? (e.g., 100% Government, 50% Industry/50% Government, etc.)

The response to this part of the question is tabulated below, where G and I refer to Government and Industry, respectively, and numbers tabulated represent percentage of the category making the selection.

Category	100%I	50%I/50%G	25%I/75%G	10%I/90%G	100%G
Academic	--	46%	12%	4%	15%
Shipbuilding	--	75%	--	25%	--
Owner/Operator	--	50%	--	--	40%
Manufacturing	12%	25%	--	12%	31%
Consultants	--	50%	--	--	29%
Government	--	33%	--	33%	33%
Combined	3%	44%	4%	7%	25%

Miscellaneous comments received are as follows

Academic --

- Government must do it because Industry has no visibility or unity.

- Government should fund 100% including prototype; all follow-on's should be funded by Industry.

Manufacturing --

- Industry can't now afford to pay for its own ships - never mind R&D.

- Incentives for Government are balance of payments & military sea transport.

- Government should fund R&D with Industry furnishing test facilities for first-of-a-kind development.

Consultants --

- Industry should foot the bill, with Government help only as needed.

Question 5 Part a) What type Government construction subsidy system (if any) do you consider should exist for nuclear ships in the U.S.?

The response regarding construction subsidy for nuclear ships is tabulated below, where the numbers represent percentage of the category stating the view.

Category	None	Yes	Same as conventional	More than conventional*
Academic	15%	23%	15%	27%
Shipbuilding	--	25%	--	75%
Owner/Operator	--	50%	30%	20%
Manufacturing	25%	25%	12%	25%
Consultants	7%	36%	29%	14%
Government	--	--	--	100%
Combined	12%	29%	18%	29%

*Miscellaneous comments, including those in this group, are as follows:

Academic --

- 80% first 10 ships, 70% next 10 ships, and 60% thereafter

- R&D and demonstration ships, plus centralized East-West-South port fuel exchange facilities

- Any building costs that turn out to be higher than expected

- Enough to break even vs. the most economic competition

- Should subsidize yards to develop capacity

to build & service nuclear ships

- Only as needed - depends on complete economic competitiveness of nuclear ships

- Decreasing after first ships

- None after first ships

<u>Shipbuilding</u> --

- Same as conventional, plus cost of first reactor including non-recurring engineering costs

- Provide incentive program to help fund nuclear vs. conventional cost differences; should reduce with time

- 50% of capital cost

<u>Owner/Operator</u> --

- Same as conventional, plus underwrite nuclear liability insurance

- Same as conventional, plus cost difference between nuclear and conventional

<u>Manufacturing</u> --

- Same as conventional, plus incentive for risk reduction on first 3-10 ships to get the program moving

- Same as conventional, plus development incentive to be repaid from operational savings

- Minimum required to maximize benefits to U.S. such as balance of payments, national security & access to raw materials, while affording economic viability

- Somewhat more than for a conventional ship

Consultants --

- Set up a government office to assist industry in overcoming government regulatory problems -- Government is one of the biggest impediments to nuclear marine propulsion

- 50% capital cost

- Not over 25% - if it can't be done for this, then now is not the time

Government --

- Same as conventional, plus difference between nuclear and conventional

- Same as conventional, plus incentive for initial "demonstration" ships

- Same as conventional, plus incentive for first application to each ship type

Question 5. Part b) What type Government operating subsidy system (if any) do you consider should exist for nuclear ships in the U.S.?

The response regarding operating subsidy for nuclear ships is tabulated below, where the numbers represent percentage of the category stating the view.

Category	None	Yes	Same as conventional	More than conventional*
Academic	35%	4%	12%	31%
Shipbuilding	--	25%	--	50%
Owner/Operator	20%	40%	10%	10%
Manufacturing	50%	19%	6%	12%
Consultants	14%	28%	28%	7%
Government	33%	--	--	67%
Combined	30%	18%	12%	22%

*Miscellaneous comments, including those in this group, are as follows:

Academic --

- Operator training and pay on first ships
- Nuclear liability insurance
- Entire fuel cycle and fuel handling at first to avoid introduction of unsafe practices
- Any operating costs that turn out to be higher than expected
- Enough to break even vs. the most economic competition
- Only as needed - depends on complete economic competitiveness of nuclear ships
- Assistance in finding solutions to labor problems
- As required by economic and foreign policies of U.S.A.

- Don't know - the present system is not working

- No more than presently "enjoyed" - if nuc's are uneconomical without subsidy, then back to fossil fuels

- None - free market competition could hopefully force more optimum utilization of manpower & general operating procedures

- Decreasing with operation

- None after first ships

Shipbuilding --

- Nuclear fuel, plus reactor plant maintenance

- Incentive to help fund nuclear vs. conventional cost differences

- 35%

Owner/Operator --

- Same as conventional, plus underwrite nuclear liability insurance

Manufacturing --

- Minimum required to maximize benefits to U.S. such as balance of payments, national security & access to raw materials, while affording economic viability

- Same as conventional, but return to government part of fuel savings

- Depends on manning required by government regulations

- Strong laws to prevent union featherbedding of manning schedules

Consultants --

- Government guarantee of berth for refueling and a disposal system for radioactive waste at a firm cost

- Must have if U.S. labor keeps insisting on more pay for less work

- Only if subsidize central stations

- If nuclear can't be made competitive without subsidy, then back to the drawing board for refinement before building more ships.

Government --

- Same as conventional, plus special port entry costs and extra licensing costs

- Same as conventional, plus incentive for first application to each ship type

Question 6. What do you consider the major reason(s) for lack of nuclear propelled commercial ships in the U.S. today? (e.g., plant capital cost (Cap Cost), operating costs (Oper Cost), labor relations (Labor Rel'ns), international liability (Liab), emotional disfavor (Emot), safety concern (Saf), plant reliability (Plant Rel'y), refueling outage time (Out Time), etc.)

The response to this question is tabulated below, where possible; other responses follow the tabulation. The numbers in the table represent the percentage of respondants in that category stating that view.

Category	Cap Cost	Oper Cost	Labor Rel'ns	Saf/ Liab	Emot	Plant Rel'y	Out Time
Academic	58%	38%	50%	27%	23%	11%	11%
Shipbuilding	100%	100%	50%	50%	25%	25%	--
Owner/Operator	70%	50%	30%	10%	--	--	20%
Manufacturing	44%	31%	12%	--	--	--	--
Consultants	50%	50%	65%	14%	21%	--	7%
Government	--	--	33%	--	--	--	--
Combined	55%	43%	41%	19%	14%	6%	8%

Responses not so amenable to tabulation, with the number (in parentheses) of times the response was made, if more than one, are as follows:

Academic --

- The major reason is a lack of U.S. commitment to a viable, competitive merchant marine, plus utter surrender to union demands. Nuclear ships can and will be built by foreign nations who have control of the above factors, when they catch up to our technology.

- Lack of government interest and support. In 1968 Sea Land approached the AEC about making the SL-7 nuclear powered, and met with a completely negative response.

- Credibility gap - too much crap published in the past

- Natural conservatism of the industry

- The industry is too debilitated to invest money in long term future development items.

- Unrealistic design and use

- Lack of profit incentive compared with large investment required and high risk of financial loss involved(6)

- SAVANNAH experience has delayed confidence and high expectations from nuclear ships.

- High maintenance and repair cost

- High liability insurance cost (2)

Shipbuilding --

- Main reason is that return on investment does not compare favorably with fossil fueled ships in U.S. without strong government subsidy.

- Nuclear liability insurance costs (2)

Owner/Operator --

- Lack of government support

- Large number of engineering personnel required

- No commercial control over nuclear fuel supply

- Excess government red tape and regulatory complications (3)

- Risk of financial loss too great (3)

- Liability insurance cost (2)

Manufacturing --

- Not sufficient economic incentive, plus government attitude. Safety concern has been used as a reason when it shouldn't have been; SAVANNAH's safety record was excellent.

- Not enough Russian commercial nuclear ships -- when they get them, we'll get them.

- SHP requirements and fuel oil cost was not high enough so no economic incentive until the last 2 years.

- A very complex issue - no one answer can do

- One man - Adm. Rickover

- No clear-cut advantage of nuclear over conventional propulsion for any present marine transportation system (2).

- Current incentives are outweighed by high financial risks involved (3).

- Unenlighted maritime unions, than which there hardly ain't no worse.

- SAVANNAH proved feasibility, but entire maritime picture regressed during her life. The Edsel has a maritime parallel.

<u>Consultants</u> --

- Lack of a "champion" in government to encourage shipping interests

- AEC requirements are abysmally restrictive

- Lack of a clear need for high SHP on a "hard" mission

- No economic incentive in the face of high financial risk and fear of losing one's shirt (3)

- International liability insurance cost (2)

Government --

- Owner concern over licensing and port entry red tape may hamper economic considerations.

- Long delivery time of some reactor items (e.g., reactor vessel) discourages owners.

- Naval Reactors

- Inadequate government industry initiative

- Regulatory uncertainty

Question 7. Any further comments you might like to make regarding commercial nuclear ship propulsion would be appreciated.

The comments received, by categories, are as follows:

Academic --

- Nuclear ship propulsion, like the Great Eastern in its day, is clearly an idea whose time has not yet come. I'd guess it'll be 1980-1990 before it does.

- SAVANNAH experience should not discourage you -- no other U.S. flag, break-bulk cargo ship was making money at the time, either.

- The environmentalists and the AEC are competing to see who can slow progress the most.

- A more urgent need is to develop central station power plants.

- I would like to see the Navy develop nuclear capability in KA type ships. I'm very pessimistic about our

merchant marine ever becoming interested and willing to spend its own money.

- Commercial nuclear ship propulsion does not merit a high position on our list of national priorities.

- Nuclear propulsion makes out where sustained periods at high SHP make nuclear fuel costs able to offset higher other nuclear costs, and where its inherent advantages can be capitalized upon: submerged operation & pollution-free operation.

- I suspect that as oil becomes depleted, artificial (e.g., made from coal) fossil fuels may be the alternative superior to nuclear.

- I believe it worthwhile to build 1-3 fast nuclear ships and support them 5-10 years; either a need will be apparent or it will flop. If the former, commercial interests can take over.

- SAVANNAH was a technical success, a practical failure. Perhaps a 2,000 ton oceanographic survey vessel should be tried next - no union problems, and government funds build it anyhow. In Alaskan waters, high fuel costs and few fueling harbors might might make such a nuc vessel valuable.

- SAVANNAH and NERVA experiences are analogous -- very successful reactor development, but no apparent mission for it.

- Going toward nuclear powered air cushion freighters would fill a void in the speed-cost performance map.

- The Navy's excessive classification of its PWR technology is a disservice to the nation and has been a major factor in the failure of our attempts toward a nuclear maritime industry. Even other defense efforts have suffered from the arrogant preoccupation of Naval Reactors with classification.

- U.S. ship construction and operation are not competitive. U.S. nuclear ship construction and operation will also be not competitive. U.S. shipyards and operations are poorly managed compared to foreign technology and operations.

- We should build enough of a nuclear fleet to bear a minimum required capacity in a national emergency, but I feel strongly that nuclear marine plants must stick close to central station technology, thus obtaining benefits of scale far beyond their own market, particularly in the nuclear fuel cycle and in some component development.

- Well coordinated knowledge, a considerable amount of effort, and dedicated personnel could realize this dream.

Shipbuilding --

- Before nuclear ship propulsion can ever be commercially competitive, either compared with conventional

U.S. or with foreign ships, the currently vast proliferation of regulatory constraints on the commercial marine industry must be realistically reappraisedand reduced.

- Development programs should be undertaken only if they would clearly produce economically competitive plants.

- If a ship operator/reactor builder/shipyard team would present a strong proposal to MarAd, backed by commitments to spend their own funds along with MarAd funds, nuclear power could be at sea in a commercial ship between 1980 and 1985.

Owner/Operator --

- I estimate it'll be at least the year 2000 before nuclear propulsion gets any real impetus

- Future prospects should be good due to Navy experience lessening concern over a nuclear accident

- A major technological breakthrough, resulting in lower hardware costs, is needed to ever make nuclear ships attractive to a commercial operator.

- I do not expect to see any economical nuclear commercial ships in my projected lifetime of 33 remaining years.

- The financial risk of nuclear ships is too high for ship operators to accept.

- Government involvement should be limited to encouragement of nuclear ship development and funding of R&D.

- We should not start a nuclear propulsion program for national prestige alone; if not economically competitive, don't build them.

Manufacturing --

- The Navy, even with its obvious requirements for speed, range, independent operation capability, etc., has had poor luck in justifying surface nuclear ships - how possibly can commercial, at least in the near future?

- Fuel oil costs should rise enough in the next several years to make large nuclear ships economically attractive. Whether the U.S. can compete any better with nuclear than it now does is doubtful unless the whole U.S. maritime industry is reorganized.

- Nuclear propulsion is cleaner, but real ecological advantages must be weighed against potential hazards due to sloppy operators, poor maintenance, etc. It is much too sophisticated to be applied to all operators, or in fact to any merchant shipping lines at this time. Unless a 100% fool-proof nuclear plant can be developed, I don't think it should be put to merchant use; also, radar and other sophisticated gear must be required to reduce the danger of collision, grounding, etc.

- The technology is here - now need to optimize to get the most economically viable ship. International aspects of licensing and nuclear liability should also be looked at more closely.

- A bold venture into LASH and SEABEE type ships is needed in nuclear power - but it must be justified by meaningful transportation system studies.

- It will have to come.

Consultants--

- Economics is all important - prove economic competitiveness to shipowners and they'll go nuclear (2).

- Only nuclear subs were designed for reactors; SAVANNAH and OTTO HAHN are normal ships fitted with reactors; the carriers are multi-reactor abortions. Current government sponsored studies for maritime applications are stifled by prior government definition of the solution (e.g., why not nuclear catamaran containerships? or why not a fleet of pulp carriers which, when in port, power the mill with the same plant that powers the ship at sea?)

- Must have labor union agreement before construction, or shipowners are not interested; the marine industry is governed by fear of loss - not expectation of profit.

- The alternative to government subsidy of any U.S. maritime effort is to allow private builders and operators to close their doors in the face of foreign competition.

- Nuclear propulsion will eventually "fly", but it may have to wait for the world's oil reserves to get much lower.

- Although gas turbines have the spotlight now, nuclear power will lead in the 90's.

- Time is on the side of nuclear, but it won't go without a government program.

Government --

- Nuclear propulsion is dead until owners feel they won't be bogged down in new construction delays, regulatory aspects in construction and operation, and labor relations problems; these hit the owner in the pocketbook even the face of higher fossil fuel costs. These areas are the ones which should receive the R&D, vice the technological areas.

- Build into the design the ability to do quick, inexpensive inspections of the pressure boundary as in central stations.

- Nuclear power for U.S. merchant ships is the only way to compete with foreign maritime - a major national benefit (2).

- If industry's reluctance to accept new technology risks can be overcome by government financial "incentive", should see nuclear ships by late '70's.

Thesis
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Analysis of past, present and future applications of nuclear power for propulsion of marine vehicles.